DECOMMISSIONING PLAN

Californium Neutron Flux Multiplier (CFX)

NRC License: SNM-1513
Docket 7001703

Kodak

EASTMAN KODAK COMPANY
1999 Lake Avenue
Rochester, New York 14650,

Revision 1

Submitted: May 2008

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1 INTRODUCTION

1.1 PURPOSE

1.1.1 The purpose of this Decommissioning Plan is to establish the management and technical protocols that Eastman Kodak Company Research Laboratories (EKC) will follow in addressing the final release of the portions of Building 82 at the EKC Activation Analysis Facility that were used to house and operate the Californium Neutron Flux Multiplier (CFX) system.

1.1.2 Since the CFX itself, related shielding materials, and the nuclear fuel have already been removed under the terms of License Amendment 3, this document will focus on final release of the CFX underground cavity from the restrictions of Nuclear Regulatory Commission (NRC) License SNM-1513. The objective is to terminate the license and to release the facility for unrestricted use.

1.2 LICENSEE IDENTIFICATION

1.2.1 EKC, whose principal address is 343 State Street, Rochester, New York 14650, holds NRC Special Nuclear Material License SNM-1513 issued in 1975 for the operation of the CFX. The license is assigned NRC docket number 7001703. The license was amended in 1989 (Amendment 1), in 2002 (Amendment 2), and in 2007 (Amendment 3). Amendment 3 was issued to permit the disassembly of the CFX and removal and disposition of the fissile material. The expiration date of the current license is September 30, 2008.

1.3 BACKGROUND

1.3.1 EKC Research Facility, Building 82 was established in 1966 to provide a dedicated research and development facility for conducting chemical and radiological analyses, doing small quantity bench and batch scale research and development studies on manufacturing processes, and to investigate new chemicals of interest to the Corporation's various operating divisions. In this facility, EKC conducted R&D work utilizing the CFX to investigate various chemical irradiations. Work with enriched uranium fuel plates in the CFX required that EKC have a Special Nuclear Materials license.

1.3.2 EKC used Special Nuclear Material (SNM) consisting of enriched uranium clad in aluminum alloy. The sealed source material was manufactured in the form of Materials

Testing Research (MTR)-type fuel plates which have been removed from the facility and returned to the Department of Energy (DOE) for final disposition or disposal.

1.3.3 A Californium-252 (Cf-252) source supplied additional neutrons which drove the CFX assembly to a sub-critical K_{eff}[1] of 0.99 and supplied a steady stream of high energy and thermal neutrons for research purposes. The Californium source was licensed by the State of New York under Kodak's license number 1347-0255. It was permanently removed on July 12, 2006 and the license inventory has been altered to reflect the change.

1.3.4 EKC managed work force safety and material accountability in accordance with the Company's Radiation Protection Program. The Radiation Safety Officer (RSO) is responsible for conformance to the Plan's requirements.

1.3.5 Typical dose monitoring results, using film badges, routinely showed occupational exposures to be less than 25 mRem/month (non-detectable) for those individuals authorized to work within the immediate vicinity of the CFX. Routine monitoring of laboratory and work areas documented that surfaces were less than 1,000 dpm/100cm^2 for tritium and 100 pCi/100 cm^2 for other β-emitters. These routine monitoring results demonstrated that activities within the vicinity of the CFX were performed in a manner that maintained exposures to radioactive materials ALARA[2] and did not result in contamination of work areas or personnel. There were never any instances where SNM contamination was encountered.

1.3.6 The CFX was dismantled and the fuel plates were packaged and shipped to the Department of Energy (DOE) Savannah River facility near Aiken, GA in November, 2007 under the terms of License Amendment 3. All of the structural parts and shielding materials from the CFX have been surveyed and either released or disposed of as Low Level Radioactive Waste (LLRW). Only the concrete cavity and the access labyrinth beneath the basement level outside the NW corner of Building 82 remain to be released.

1.3.7 A Radiological Characterization Survey was conducted after removal of all the CFX materials from Nov, 2007 to February, 2008 in order to provide criteria for the Final Status Survey and final release of the facility. The results of the Characterization Survey have been published separately in the Radiological Characterization Report (RCR).[3]

1.3.8 EKC no longer uses the licensed material, and the facility is therefore being decommissioned to allow for license termination and unrestricted release. EKC decommissioning activities that remain include the removal of any residual activity that may be required[4], and the survey and release of the underground cavity and access labyrinth where the CFX was used. When completed, the facility will meet all applicable unrestricted release criteria.

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1 K_{eff} is the effective neutron multiplication factor - defined as the number of neutrons in a generation divided by the number of neutrons in a preceding generation. Values of K_{eff}=1 indicate that a system is self sustaining, values in excess of one indicate that a system is super-critical.

2 As Low As Reasonably Achievable

3 Eastman Kodak Co., *CFX Radiological Characterization Report*, NEXTEP Consulting Group, Inc., April 2008.

4 Depending upon the approved release limits.

1.3.9 Decommissioning will be performed under written policies and procedures, incorporating appropriate methods outlined in NUREG-1757[5] as required. In accordance with NUREG-1757 criteria, EKC decommissioning activities fall under the Group 3 facility requirements since some of the surrounding materials have been activated by neutron bombardment during the course of normal operations.

[5] In September 2003, the U.S. Nuclear Regulatory Commission (NRC) published a three-volume NUREG report, NUREG-1757, "*Consolidated NMSS Decommissioning Guidance*." NUREG-1757 provides guidance on: planning and implementing license termination under the License Termination Rule (LTR) in 10 CFR Part 20, Subpart E; complying with the radiological criteria for license termination; and complying with the requirements for financial assurance and recordkeeping for decommissioning and timeliness in decommissioning of materials facilities.

2 FACILITY INFORMATION

2.1 SITE AND GROUNDS

2.1.1 EKC's Research Facility, Building 82 is located within the EKC Research Complex. The Research Complex is located in Monroe County and is within the City limits of Rochester, New York, approximately 3.5 miles from the downtown district. The Research Complex comprises approximately 49 acres of land, with facility buildings encompassing approximately 15% of the facility grounds. The rest of the land is covered primarily by parking lot pavement. An overhead photograph, showing the EKC Research Complex and grounds in relation to the major roads that comprise its boundaries, is provided in Figure A-2.1.[6]

2.1.2 The research buildings are situated on the southeastern part of the property with the parking lot area situated to the north and west of the facility buildings. The Research Complex is bordered by Eastman Avenue on the south, Lake Avenue on the east, Merrill Street on the north, and Goodwill Street on the west. The immediate environs surrounding the research buildings include a mix of industrial and residential use. The EKC Kodak Park manufacturing site is located to the south of the Research Complex. Residential housing is located to the west and north of the property. The Genesee River, flowing in a northerly direction, is located approximately 300 meters to the east of Lake Avenue.

2.1.3 EKC CFX activities were confined to an underground cavity located at the northwestern corner of Building 82 within the Research Complex. An aerial photo of the EKC research buildings is provided in Figure A-2.2, and it shows the approximate location of the CFX cavity and control room.

2.1.4 Access to the buildings and the CFX cavity are secured by several redundant access controls.

2.2 RADIATION CAVITY

2.2.1 Laboratory Description

2.2.1.1 Special nuclear material was only used within the CFX assembly located in the radiation cavity of Building 82. The cavity is underground, outside but adjacent to the northwest corner of Building 82 and below basement floor level. Figure A-2.2 shows the location of the CFX in the aerial photo of the facility. A diagram of the cavity and control room floor plan is presented in Figure A-2.3.

2.2.1.2 The walls of the cavity are made of two-feet thick, high-density, poured concrete. The floor of the cavity is one-foot thick poured concrete resting on bedrock. The cavity ceiling is poured concrete two-feet thick with nine and one-third feet of earth above it. Seven and one-half feet (measured in a horizontal plane) of earth fill lie between the cavity wall and the outer west wall of the control room located in the basement of Building 82. The shortest distance between the cavity ceiling and control room floor is

[6] All figures or tables beginning with an alphabetical character are included in the corresponding appendix. The appendix designation will be left out of future references to such figures and tables.

eight and one-half feet through earth fill. The dimensions of the cavity are 15ft. x 24ft. x 9 ft. high. Access to the cavity is via a circular stairway. The entrance to the cavity is closed off by a folding, accordion-type, steel limited access gate with a lock which is part of the radiation cavity Safety Interlock System.

2.2.1.3 Starting in the mid-1970s, two 14 MeV neutron generators were added to the radiation cavity and they were also used for R&D purposes. In 1999, one of the generators was reconditioned for further planned research activities. The other generator is currently inoperable. The neutron generators and the portion of the cavity occupied by them will not be released as part of the CFX project.

2.2.2 Radiation Cavity Safety Interlock System

2.2.2.1 The Safety Interlock System was established to prevent access to the radiation cavity when unsafe conditions prevailed. It was designed to prevent access under the following conditions.

- The 14 MeV Neutron Generator was energized with high voltage
- The radiographic shutter of the CFX was open and the CFX was operating
- The level of exposure within the radiation cavity exceeded the threshold level of 10 mR/h.

2.2.2.2 The safety interlock system consists of two linked sub-systems, one primarily for the 14 MeV generator, the other controlling the operation of the CFX. The latter has been deactivated since removal of the CFX.

2.3 CFX DESCRIPTION

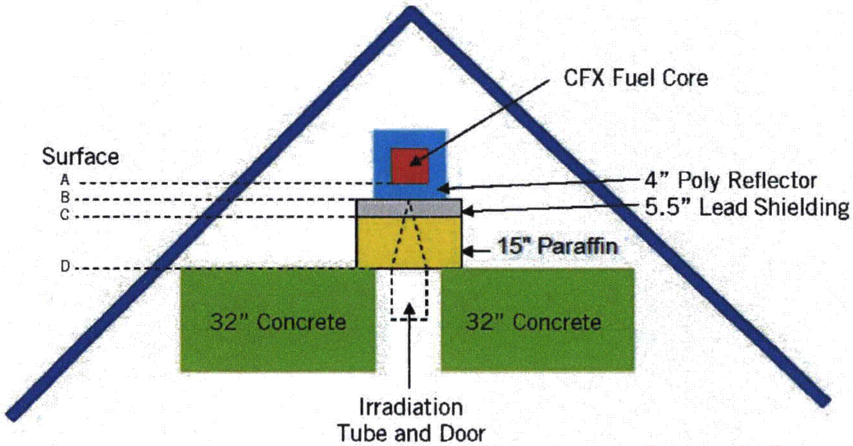
2.3.1 General Information

2.3.1.1 In 1974 EKC purchased the CFX from Intelcom Rad Tech of San Diego, California. It is one of only two such devices ever produced. The second was installed at the DOE Mound facility in Ohio and was decommissioned in the 1990's.

2.3.1.2 The CFX was a sub-critical assembly of uranium-235 surrounding a Cf-252 source. The function of the U-235 fuel was to multiply the neutrons coming from the Cf-252 source, which fissions spontaneously. The CFX was designed never to exceed a K_{eff} of 0.99. The CFX assembly yielded sufficient neutron fluxes for applications such as neutron activation analysis.

2.3.1.3 Details and characteristics of the CFX are presented in the EKC CFX Scoping Study[7]. A diagram showing the arrangement of the CFX fuel assembly and the surrounding shield materials as they were arranged in the cavity prior to removal is presented in Figure 2.1.

[7] NEXTEP TM0703, *Scoping Study, CFX Decommissioning Project, Eastman Kodak Company*, R. Newman and Ning Zhang

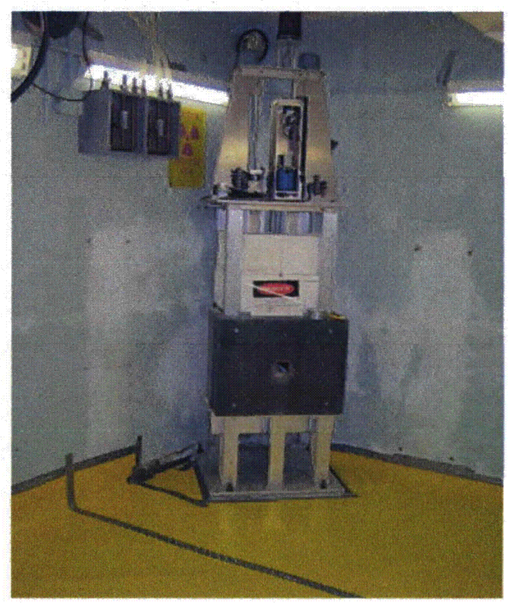


CFX Operational Configuration
Figure 2.1

2.3.2 The CFX was dismantled in November, 2007 and removed from its mountings in the corner of the sub-basement labyrinth located in the Kodak Research Complex in Rochester, NY. A picture of the CFX with all the shielding materials removed is presented in Figure 2.2.

2.4 CFX Operating History

2.4.1 The CFX became fully operational in 1975 for neutron activation analyses and for the testing of recording materials for neutron radiography. The Cf-252 source is believed to have remained inserted into the CFX for most of the life of the CFX such that the total estimated operational hours are 252,000 at an average power output of 5.8 watts. The original 1.0 mg Cf-252 source was replaced on four occasions after startup. Replacements occurred in 1977, 1983, 1990 and 1997. Each of the Cf-252 replacement sources were 3.1 mg, larger than the original 1.0 mg source at startup. The CFX was permanently removed from service in June, 2006, and the Cf-252 source was removed from the CFX on July 12, 2006.



CFX Structure and Mount
Figure 2.2

2.4.2 The CFX assembly was never moved from its original setup location until de-fueling took place in November, 2007. Routine leak test surveys were performed throughout the operating history with no indication of nuclear material contamination.

2.4.3 Routine Operational Radiological Surveys

2.4.3.1 Monthly radioactive surface contamination checks were performed in the radiation cavity to ensure that radiological contamination had not occurred. Designated areas were wiped with paper disks to sample removable radioactive contamination and then counted for β-activity in liquid scintillation counters. The removable radioactive contamination never exceeded the administrative limits of 1,000 dpm/100cm^2 for tritium or 100 pCi/100 cm^2 for other β-emitters.

2.4.3.2 Semi-annual wipe samples were collected from the interior of the CFX assembly, accessible through the radiographic port and tested for α-activity by scintillation or gas proportional counting. 500 pCi of removable α-activity, the level presented as SNM-1513, Amendment 2, Condition 11, was never encountered. All wipe testing was performed in accordance with the license.[8]

2.4.3.3 The last leak test was performed on September 19, 2006. The wipe sample was collected through the radiographic port and the result (<0.38pCi gross alpha) was well below the threshold (500 pCi) established for the test, demonstrating that the sealed sources within the CFX were still intact.

2.4.4 Spills

2.4.4.1 Due to the nature of the sealed sources and the evidence of no leakage, no radiological spills were encountered at the EKC research facility as a result of the CFX handling or operation.

2.4.5 On-Site Disposal

2.4.5.1 No licensed or activated materials were ever disposed of on site. The MTR fuel plates were removed under DOE supervision in November of 2007, and they were transferred to the Savannah River facility near Aiken, SC for final disposition. Structural components and shield materials showing radioactivity above background or products of activation above exempt levels were disposed of at a licensed LLRW facility.

2.5 Previous Decommissioning Activities

2.5.1 On July 17, 2006, a formal notice was sent by EKC to the NRC that EKC had ceased normal CFX operations/activities as specified in special nuclear material license SNM-1513.

2.5.2 The Cf-252 source was removed and shipped off-site on July 12, 2006, rendering the assembly inoperable.

2.5.3 On 9 February 2007, EKC submitted application for License Amendment 3 in order to obtain permission to disassemble and de-fuel the CFX. The amendment was granted November 21, 2007.

[8] SNM-1513 Attachment, "*License Condition for Leak Testing Sealed Uranium Sources*", April 1993.

2.5.4 De-Fueling

2.5.4.1 The CFX was disassembled and de-fueled from 26 to 30 November, 2007 in accordance with the De-Fueling Work Plan[9] and the Shipper/Receiver Agreement (S/RA).[10] The MTR fuel plates were returned to the DOE for disposal. The shield materials and disassembled components of the CFX were surveyed and released or disposed of as LLRW.

2.5.4.2 Radiological surveys performed in connection with the de-fueling process were recorded and documented in NEXTEP Technical Memorandum (TM) 0714.[11] All the requirements of the DOE as expressed in the S/RA were met and the material was accepted for shipment. All of the structural components and shielding materials were surveyed and either released or disposed of in accordance with the terms of License SNM-1513. Surveys of the cavity during de-fueling revealed no measurable contamination from licensed materials.

2.5.5 Characterization Surveys

2.5.5.1 After completion of the de-fueling operation, the Characterization Survey was taken of the cavity and the labyrinth in accordance with the CFX Characterization Plan.[12] Supplemental surveys were also performed to augment the data set during the period from December, 2007 through February 2008. The results are presented in the Radiological Characterization Report (RCR)[13] which was published in April, 2008.

2.5.5.2 The Site Characterization Surveys discovered no evidence of contamination with licensed materials or fission products. A portion of the south and west walls and the floor of the CFX cavity showed evidence of neutron activation which remained below the proposed release limits for activated material.

2.5.6 Sumps Pumps/Drain Lines/Sewer

2.5.6.1 All licensed special nuclear material was contained within the sealed source fuel plates inside the CFX fuel assembly, and no evidence was found during the defueling operation or the Characterization Survey of any contamination which might indicate that the protective cladding around the fuel in the MTR plates had been breached. Because of this, removal of the sump pump, drain line, or sewer systems in the radiation cavity in search of residual contamination was not deemed necessary.

2.5.7 Air Ventilation System

2.5.7.1 The existing cavity air ventilation system was monitored for removable alpha activity greater than 200 dpm/100cm^2 during operations and de-fueling with no significant findings. Air samples collected during the de-fueling process revealed no detectable levels of airborne radioactivity. Surveys have not shown accumulations of radioactive

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9 Eastman Kodak Company, *CFX De-Fueling Work Plan*, Rev. 1, NEXTEP Consulting Group, Inc., November 2007.

10 Washington Savannah River Co./Eastman Kodak Co., *Material Control and Accountability Shipper/Receiver Agreement between Washington Savannah River Co. and Eastman Kodak for Highly Enriched Uranium Material,* July 23, 2007 as amended by Addendum A, November 17, 2007.

11 NEXTEP TM0714 – *Kodak CFX De-fueling Documentation*, Barton P. Anderson.

12 Eastman Kodak Co., *CFX Characterization Plan,* NEXTEP Consulting Group, Inc., Nov. 2007.

13 Ibid.

material on ventilation ducts, and EKC does not expect to encounter any special nuclear material in these locations when the final status surveys are performed.

2.5.8 **Activated Materials**

2.5.8.1 The long operational history of the CFX has activated some of the surrounding concrete producing Co-60, Eu-152 and Eu-154 radionuclides as discussed in the RCR. Activated structural components or shield materials in excess of exempt quantities were disposed of as LLRW.

2.6 GROUNDWATER AND SURFACE SOIL

2.6.1 The authorization and use of SNM per NRC license SNM-1513 precluded the possession and use of material other than the sealed uranium sources in the CFX fuel assembly. As evidenced by routine smears and surveys, the sources have remained intact during the CFX operating history. No evidence of contamination within the concrete cavity exists, and no on-site disposal has occurred. Therefore groundwater and the surface soils surrounding the concrete CFX cavity are deemed unaffected by CFX operations.

3 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

3.1 DECOMMISSIONING OBJECTIVE

3.1.1 The decommissioning objective is to return the Building 82 radiation cavity to levels that meet the NRC's unrestricted use criteria, and to terminate Special Nuclear Material License SNM-1513 in accordance with NUREG-1757 requirements for a Group 3 facility. The specific area of concern relative to license decommissioning is the portion of the radiation cavity that was in close proximity to the CFX during its operating life. (See figure 2.1).

3.1.2 The objective includes the survey and, if required, decontamination of portions of the radiation cavity. Radioactive materials left in place, if necessary, will be present in ALARA concentrations such that any individual member of the public will receive a total annual dose of less than 25 mrem (TEDE[14]) above background from all sources.

3.2 PLANNED DECOMMISSIONING ACTIVITIES

3.2.1 After criteria for unrestricted release have been approved, any necessary remediation will be performed, and a Radiological Final Status Survey (FSS) will be completed and submitted for final approval.

3.2.2 The FSS of building surfaces and fixed equipment will be planned and conducted utilizing techniques and sampling protocols as described in Section 6. The surveys will account for the possible presence of licensed material, the associated products of fission and decay, and neutron activation of materials surrounding the CFX.

3.2.3 Wherever possible, data already collected as part of the Characterization Survey which meet the quality criteria for FSS data will be used to satisfy the requirements of the Final Status Survey.

3.3 RADIONUCLIDES OF INTEREST

3.3.1 Contamination with Licensed Materials

3.3.1.1 NRC license SNM-1513 authorizes the possession and use of up to 1,582.6 g of enriched U-235 in the form of MTR Type fuel plates. During the operating history of the CFX, the licensed material remained in place within the CFX fuel assembly. After 250,000 hours of continuous operation at 5.8w power, approximately 0.074 g of U-235 were consumed by fission.[15]

3.3.1.2 Modeling of the residual activity and fission product buildup was performed by the SCALE[16] system, and is described in detail in NEXTEP TM0519[17]. A Listing of the

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[14] Total Effective Dose Equivalent: a term for total dose received from radioactivity. It consists of the sum of the Deep Dose Equivalent (DDE) for external exposure, and the Committed Effective Dose Equivalent (CEDE) for internal exposure.

[15] NEXTEP TM0711 *Kodak CFX: Estimated U-235 burn-up and Residual Mass,* Bobby Leonard, PhD and Andrew H. Thatcher, CHP.

[16] *SCALE:* A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations, ORNL/TM-2005/39, Version 5, Vols. I-III, April 2005.

[17] NEXTEP TM0519, *Core Residual Activities and Dose Rates From the Kodak CFX Facility,* A.H.Thatcher, CHP

estimated residual activity from U-235 and its progeny is presented in Table B-3.1, and the estimated residual activity from fission products is presented in Table B-3.2.

3.3.1.3 This analysis showed that the activity from fission products would dominate the mix of any contaminants found in the CFX cavity[18]. Moreover, among the products of fission, Sr-90, Cs-137 and their progeny dominated by at least one order of magnitude, and were present in approximately equal quantities.

3.3.1.4 Since any contamination found within the CFX facility after disassembly and removal of the fuel assembly would be from the rupture of one or more of the MTR-type fuel plates, the resulting contamination would contain a mixture of licensed U-235 and its natural decay products as well as the products of fission.

3.3.1.5 Since Cs-137 and Sr-90 are both strong beta emitters, the primary means for detecting evidence of contamination from the CFX is beta detection by direct measurements and scans.

3.3.2 Induced Radioactivity due to Neutron Activation

3.3.2.1 Radiological activity above background in building surfaces and equipment may also come from induced radioactivity caused by neutron activation. A theoretical analysis of the likely results from 30 years of activation is presented in TM0520[19]. According to that analysis, it is reasonable to expect that the only activation of significance would be that caused within the surrounding concrete and embedded reinforcing bars in the SW corner of the cavity where the CFX was mounted.

3.3.2.2 After de-fueling and during the Characterization Surveys, the concrete pad[20] on which the CFX structure was mounted was sampled and measured in a low background Germanium detector system to identify the radionuclides present in the concrete from neutron activation. The results are presented in the RCR and are tabulated in Table 3.1.

Table 3.1
Concrete Activation Analysis

Isotope	Normalized Activity
^{60}Co	7.66 pCi/gm
^{152}Eu	14.65 pCi/gm
^{154}Eu	0.95 pCi/gm

3.3.2.3 Since all the radioisotopes shown in Table 3.1 are strong gamma emitters and, since activation products in concrete would be distributed with depth and so attenuate some of the beta emissions, gamma surveys were the primary means used to detect activation products in the concrete surrounding the CFX.

18 The total predicted activity from fission products was 330 mCi compared to the total activity due to U-235 and its daughters of 7 mCi.

19 NEXTEP TM 0520, *Neutron Activation Analysis for Kodak CFX Facility,* A.H. Thatcher, CHP

20 This mortar pad was used for leveling the CFX and had been poured on top of the cavity floor. After CFX removal, the pad was broken up and removed as well. Scan measurements on the floor of C1 were repeated in the absence of the leveling pad.

3.3.2.4 During the Characterization Survey, a portion of the south and west walls and the floor of the CFX cavity showed evidence of neutron activation which remained below the proposed release limits for activated material.

3.4 REPRESENTATIVE BACKGROUND REFERENCE AREAS

3.4.1 Surface Beta Background

3.4.1.1 Representative background areas were selected in the control room on the basement level above the labyrinth. Beta background measurements were taken on two matrix substrates: bare concrete (C) and vinyl floor tile over concrete (VT). The data are stored in the Radiation Database and a listing is presented in Table B-3.3.

3.4.1.2 Beta/gamma background for scans was computed as the average of the open window[21] measurements from the background table. These data are also included in Table B-3.3.

3.4.2 Gamma Background

3.4.2.1 Gamma background was measured in the cavity during the Characterization survey in November, 2007, and again in February, 2008 as part of a more detailed gamma survey of the activated concrete surfaces. The second survey was performed after removal of the CFX concrete mounting pad from the floor. Details are presented in the RCR. The average of three gamma background counts from the second survey was **2,791 cpm.**

3.4.3 A summary of the background data to be used for evaluation of the Characterization and Final Status Surveys is presented in Table 3.2. Materials for which no background reference measurements can be obtained will be assumed to have background values equal to zero or the LLD for the applicable instrumentation for purposes of the FSS.

Table 3.2
CFX Background Data

Matrix	β/γ Scan (cpm)	Beta Direct (dpm/100cm^2)	Gamma (cpm)
Concrete	253	580	2,791
Vinyl Tile	200	60	N/A

3.5 AFFECTED AREAS

3.5.1 The CFX cavity and underground labyrinth were divided into five constructive "rooms" for survey planning and cataloging radiological data. These rooms are pictured in Figure A-3.1.

3.5.2 Four rooms were given presumptive classifications and surveyed during Characterization and the results are published in the RCR. No contamination from licensed materials or products of fission was discovered in any of the four rooms surveyed. Radioactivity from

[21] Beta measurements were taken using an open/closed window methodology which is explained in detail in the RCR and §6 of this DPlan. Open window readings include both β and γ, while the closed window blocks the β counts. The difference in the readings is used to isolate the β counts.

activated concrete was discovered in the corner where the CFX fuel assembly was mounted in room C1. No activation was detected in any of the other survey rooms.

3.5.3 On the basis of the Characterization Survey results, the survey rooms in the CFX labyrinth have been reclassified for the Final Status Survey according to Table 3.3. The survey units are depicted in Figure A-3.2.

Table 3.3
Survey Unit Classification

Room	Description	Survey Unit	Classification
C1	Portion of the cavity which housed the CFX equipment extending 8 ft. from the corner. Includes room C1.	101	Class 2
C2	Portion of the cavity room excluding 6 ft. of the eastern end which houses the neutron generating machines and related equipment. Room C2.	102	Class 3
C3	Labyrinth Corridor (west)		Unaffected
C4	Labyrinth Corridor (north)		Unaffected
C5	Labyrinth Corridor (east)		Unaffected

3.6 RELEASE CRITERIA FOR BUILDING SURFACES & FIXED EQUIPMENT

3.6.1 Surface Contamination

3.6.1.1 The basis for release of equipment or materials is contained in Attachment 2 to License SNM-1513[22] (the Guidelines). A copy of the Guidelines is presented in Table B-3.4.

3.6.1.2 This table specifies acceptable levels of contamination for four different groups of radionuclides in terms of disintegrations per minute (dpm) for alpha and β/γ emissions, applied independently to each.

3.6.1.3 Since any contamination found within the CFX facility after disassembly and removal of the fuel assembly will be from the rupture of one or more of the MTR-type fuel plates, the resulting contamination will contain a mixture of licensed U-235 and its natural decay products as well as the products of fission.

3.6.1.4 Inspection of Tables B-3.1 and B-3.2 shows that the activity from fission products will dominate the mix of contaminants. The total activity from products of fission is 330 mCi compared to the total activity due to U-235 and its daughters of 7 mCi. Moreover, among the products of fission, Sr-90, Cs-137 and their progeny dominate by at least one order of magnitude, and they are present in approximately equal quantities.

3.6.1.5 Sr-90 and its progeny fall in the third category of nuclides listed in Table B-3.4, and they have a release limit of 1,000 dpm/100 cm^2 averaged over $1m^2$. Cs-137 and its

[22] NRC license SNM-1513, Attachment 2, *Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material,* US NRC, 1993

progeny fall into the fourth category of beta-gamma emitters and they have an average limit of 5,000 dpm/100 cm^2.

3.6.1.6 In order to identify the existence of a mixture of cesium and strontium in equal quantities, composite Derived Concentration Guideline Levels (DCGL)[23] were developed in TM0713[24] using the unity rule, and they will be applied as shown in Table 3.4.

Table 3.4
Composite Release Limits[a] for Building Surface Contamination

Nuclide	DCGLw[25] (dpm/100cm^2)	DCGL$_{EMC}$[26] (dpm/100cm^2)	Removable (dpm/100cm^2)
Sr-90	1,000	3,000	200
Cs-137	5,000	15,000	1,000[b]
Cs/Sr mix	**1,667**	**5,000**	**333[b]**

a. Above background
b. β/γ only

3.6.1.7 Since any expected contamination will be characterized by cesium and strontium, surveys will focus on beta detection (both are strong beta emitters). Beta measurements will be made using an "open window/closed window" methodology which is described in TM0713, and they will be evaluated against the composite limits in Table 3.4.

3.6.1.8 If surface contamination has been detected in excess of the composite release limits in Table 3.4, either the contamination will be cleaned up to the release criteria, or samples will be sent out to a laboratory for isotopic analysis and the limits will be applied to each isotope identified using the unity rule.

3.6.2 Activated Surfaces

3.6.2.1 The Guidelines included in the License declare that the limits (Figure B-3.4) "do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control is considered on a case-by-case basis."

3.6.2.2 To accommodate the activated materials in the CFX facility, release limits for activated surfaces were developed for this decommissioning from the isotopic analysis of the activated concrete immediately beneath the CFX. The methods and assumptions used in developing the activation release limits are presented in NEXTEP TM0604[27] which is

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23 DCGL refers to the Derived Concentration Guideline Level as defined in §2.2 of MARSSIM. It represents the predicted surface area concentration of specific radionuclides that could result in a dose equal to the release criterion.

24 NEXTEP TM0713 – *Technical Basis Document for Kodak Beta Measurements,* A.H. Thatcher, CHP (Included as Attachment E to the RCR)

25 DCGL$_W$. The "w" refers to the Wilcoxon rank sum test used to evaluate the average contaminant level over a given area.

26 DCGL$_{EMC}$ refers to the guideline used to evaluate small areas of elevated activity according to the Elevated Measurement Comparison guidelines in MARSSIM.

27 NEXTEP TM 06-04, *Kodak CFX: Development of Release Limits for Activated Surfaces and Material,* A.H. Thatcher, CHP, included as Attachment D to the RCR.

included in the RCR. The proposed limits derived for activation products in concrete are listed in Table 3.5.

Table 3.5
Proposed Release Limits[a] for Neutron-Activated Building Surfaces

Isotope	Activity (pCi/g)
Co-60	150
Eu-152	290
Eu-154	19

[a] Above background

3.6.3 The release criteria for building surfaces are also applied to fixed equipment that is not expected to be removed. The fixed equipment have the survey, sampling and release criteria of building structures and slabs. Some examples of building surfaces and fixed equipment are floors, walls, ceilings, lighting fixtures, electrical conduits, and ventilation ducts.

3.7 ACTION THRESHOLDS

3.7.1 Investigation thresholds for direct beta measurements and β/γ scans were developed and documented in TM0713[28] which is presented in Attachment E to the RCR. Derivation of a gamma threshold was developed based upon the limits in Table 3.5 and the relative abundance of each radionuclide in the concrete below the CFX.

3.7.2 The derivation of gamma conversion factors and the action threshold for activated concrete is presented in TM0801[29] which is included as Attachment F to the RCR. A summary of all the action thresholds is presented in Table 3.6.

Table 3.6
Action Thresholds

Survey Type	Net	Gross[c]	
		C	VT
Gamma (scan or direct)	360,000 cpm	363,000 cpm	N/A
Direct Beta[a]	174 cpm	235 cpm	181 cpm
β/γ Scan[b]	370 cpm	623 cpm	570 cpm

[a] Open window minus closed window (Co-Cc)
[b] Open window only (Co)
[c] Including matrix background from concrete (C) and Vinyl Tile (VT)

[28] NEXTEP TM0713, Ibid. (Attachment E to the RCR)
[29] NEXTEP TM0801 *CFX Gamma Conversion Factors and Survey Thresholds*, A.H. Thatcher, CHP (Attachment F to the RCR)

3.8 RELEASE CRITERIA FOR REMOVABLE MATERIALS AND EQUIPMENT

3.8.1 Materials and equipment that will be removed from the CFX cavity will be released based upon the contamination limits specified in Table 3.4. The contamination limits are the same limits that were specified for building surfaces.

3.8.2 **Activated Materials and Equipment**

3.8.2.1 Removable materials and equipment which have been activated by neutron bombardment have all been disposed of to a licensed LLRW facility if the levels of activation exceeded exempt quantities except for the concrete blocks which made up most of the shield wall in front of the CFX.

3.8.2.2 Release limits for the radioisotopes analyzed from a crushed sample of activated shield block were developed from Table 6.5 of NUREG 1640[30] as documented in TM0604[31]. They represent the concentration of the activated radionuclides in the concrete sufficient to produce an annual dose of 10μSv/yr (1.0 mRem/yr) to the critical group working with concrete rubble. The proposed release limits for activated concrete blocks are presented in Table 3.7.

Table 3.7

Proposed Free Release Limits for Activated Concrete Block

Element	Solid Concentration (Bq/g)	Solid Concentration (pCi/g)	Analyzed Activity (pCi/g)
Co-58	0.132	3.6	0.14
Co-60	0.033	0.9	0.33
Eu-152	0.081	2.2	0.03

[a] Above background

3.8.2.3 Table 3.7 shows the results of gamma spectroscopic analysis of a sample taken from the block with the highest level of activity in the shield.

3.8.3 Any material or equipment having contamination above the release criteria will be disposed of at a licensed radioactive waste disposal facility.

3.9 INSTRUMENTATION

3.9.1 A list of the radiation measuring equipment made available to the CFX Project is presented in Appendix C, Table C-1.

3.9.2 Gamma measurements of record were recorded during Characterization using Ludlum 44-10 2x2" Sodium Iodide (NaI) probe connected to a Ludlum 2221 Scaler/Ratemeter. A 1/8" lead shield was wrapped around the probe to reduce background noise, and all measurements were taken in the shielded configuration. The vendor specifications and description are included in Appendix C.

[30] NUREG-1640, *Radiological Assessment for Clearance of Materials from Nuclear Facilities,* Anigstein, R, Chmelynski, H.J. et al, June 2003

[31] NEXTEP TM0604, Ibid., (included as Attachment D to the RCR)

3.9.3 Direct (integrated) beta measurements and β/γ scan surveys of record were recorded during Characterization using a Ludlum 43-89 alpha/beta scintillation probe connected to a Ludlum 2224-1 scaler/ ratemeter. The vendor specifications and description are included in Appendix C.

3.9.4 These same instruments (or equivalents) will be used for the FSS.

3.10 MINIMUM DETECTABLE CONCENTRATION (MDC)

3.10.1 The MDC requirements for CFX surveys have been developed in accordance with MARSSIM[32] Chapter 4 guidelines for the primary project survey instruments.

3.10.2 MDC refers to the intrinsic detection capability of the entire measurement process. It represents the lowest level of radioactivity that will yield a net count, above system blank, that will be detected with at least 95% probability and with no greater than a 5% probability of falsely concluding that a blank observation represents a real signal. For a first pass scan the statistics were chosen to provide a 60% chance of false positive and a 5% chance of a false negative.[33]

3.10.3 The target MDC for beta direct is 50% of the composite average limit (DCGLw); for β/γ scans it is equal to the average ($DCGL_W$)[34] and ; and for gamma direct or scan it is 50% of the calculated gamma threshold from TM0801[35].

3.10.4 Details concerning the selection and calculation of MDC's for beta measurements and scans are presented in TM0713, and the derivation of the MDCR for gamma measurements is presented in TM0801. (Both are included in the RCR.) A summary comparison of the MDCs calculated for the L43-89 α/β scintillation detector and the 2"x2" NaI gamma detector with the relevant requirements is provided in Table 3.8.

Table 3.8
Minimum Detectable Concentration (MDC or MDCR) Comparison[a]

Survey Type	Target MDC(R)	Calculated MDC(R)
Direct Beta	833 dpm/100 cm^2	757 dpm/100 cm^2
β/γ Scan (Class 2)	1,667 dpm/100 cm^2	1,640 dpm/100 cm^2
Gamma (scan or direct)	180,000 cpm	910 cpm

[a] All Values given are net of background. Only the matrix with the highest value for MDC is shown.

3.11 ALARA CONSIDERATIONS

3.11.1 The greatest potential for exposure to radioactivity was during the de-fueling process which was carried out under License Amendment 3 in November, 2007. Residual activities measured in the CFX cavity during Characterization are well below all the limits for unrestricted release proposed in this DPlan.

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32 NUREG 1575, *Multi Agency Radiation Survey and Site Investigation Manual.*
33 See §4.3, TM0713
34 Class 2 and 3 survey units. For Class 1 survey units use the maximum limit ($DCGL_{EMC}$). See §3.10.
35 Ibid.

3.11.2 Although no contamination from licensed materials has been detected in the cavity, and though activation levels are very low, any decommissioning work that may yet be required will be guided by ALARA principles and substantial planning efforts will be incorporated to ensure that decommissioning activity exposures fall well within regulatory guidelines.

3.12 DECOMMISSIONING WASTE MANAGEMENT

3.12.1 No additional materials remain in the CFX cavity which require release or disposal. The fuel plates were released to the DOE facility at Savannah River, SC, and the structural components and shield materials removed during de-fueling were released according to the terms of the license or disposed of as LLRW to a licensed facility.

3.12.2 Waste Generation

3.12.2.1 Characterization revealed small areas of activated concrete in the corner of the CFX cavity where the system was mounted. If remediation of these surfaces is required, a small amount of concrete debris will be generated and disposed of as LLRW in accordance with the Kodak Radiation Protection Program. However, current plans are to let the activated concrete continue to decay in place.

3.12.3 Free Release of Materials

3.12.3.1 Materials or equipment to be free-released will be cleaned if necessary, and monitored with radiation detection instruments. All surfaces will be inspected and released in accordance with NRC free release criteria as presented in section 3.6.

3.13 CREDIBLE ACCIDENTS RESULTING FROM DECOMMISSIONING ACTIVITIES

3.13.1 EKC has evaluated the potential for exposure from conditions occurring both on-site and off-site. Since the MTR fuel plates, the Cf-252 source, and all of the CFX structural components and shielding materials have been removed and either released or disposed of, no possible accident scenario is conceivable that would produce any danger to personnel or property.

3.13.2 For activated concrete surfaces, even if the activation exceeds approved release limits, there are currently no plans to remediate the surfaces as the activated concrete can decay in place. No accident is envisioned that would result in a significant exposure to occupational workers, and it is therefore removed from further consideration.

4 DECOMMISSIONING ORGANIZATION AND ADMINISTRATION

4.1 EKC MANAGEMENT ORGANIZATION

4.1.1 The EKC Corporate Engineering and Analytical Sciences Division will have primary responsibility for the Californium Neutron Flux Multiplier and Building 82 decommissioning activities. The EKC Program Manager (EKC-PM) will be selected from this group and is responsible for overseeing and managing all decommissioning activities. The EKC-PM will report to the Director of the Analytical Sciences Center, who has management responsibility for all EKC Research Laboratories activities connected with this project.

4.1.2 Two additional Kodak personnel will be assigned to assist the EKC-PM in managing the CFX Decommissioning Project. The EKC Radiation Safety Officer (RSO) will oversee the implementation of Kodak's Radiation Protection Program and Safety Program by the CFX Decommissioning Contractor, and he will report to the EKC-PM in the capacity of Corporate RSO and Environmental Safety and Health (ES&H) Officer.

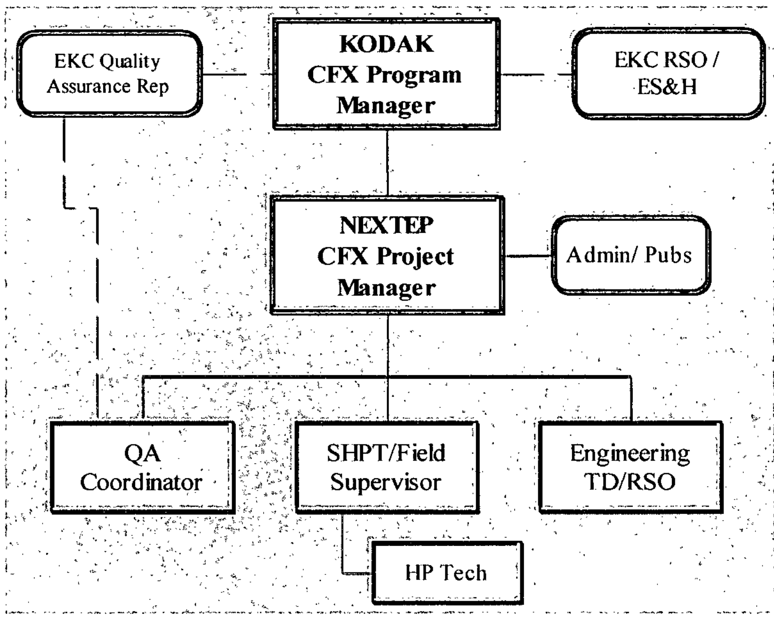
4.1.3 A Quality Assurance Representative (QAR) will also be assigned to the Project. He will report to the EKC-PM and will be responsible for reviewing all project procedures and plans for conformity and compatibility with the Kodak QA Program and Management Policy.

4.1.4 A diagram of the CFX Decommissioning Project Management Organization is presented in Figure 4.1.

4.2 DECOMMISSIONING CONTRACTOR ORGANIZATION

4.2.1 Eastman Kodak Company has obtained the services of NEXTEP Consulting Group, Inc. (NEXTEP) to assist in the planning, development, and implementation of CFX Decommissioning activities. NEXTEP, located in Louisville, Kentucky, is currently assisting EKC in carrying out the program.

4.2.2 NEXTEP has appointed a CFX Project Manager (CFX-PM) to manage the development of this Decommissioning Plan, the Work Plan, and the necessary procedures. He will also oversee all site work that may be required.



Project Management Organization
Figure 4.1

4.2.3 The CFX-PM is assisted by the NEXTEP Technical Director (TD), a Certified Health Physicist (CHP), and his technical staff. The TD will oversee all technical research and sign off on this Plan and will review all related Plans and Procedures. He will also be a primary point of contact for liaison with NRC personnel for the project and will act as the Project RSO.

4.2.4 NEXTEP's Project QA Coordinator (QAC) will ensure that adequate quality controls are established and written into the project Plans and Procedures, and he will review all controlled documents.

4.2.5 The QAC will conduct a readiness review just before on-site work begins to ensure that everything has been completed in accordance with approved plans and procedures. During site operations he will conduct audits and surveillances as required to ensure that work is performed in accordance with procedures, and that data conform to the data quality objectives. He is responsible to ensure that personnel are properly trained, documentation is kept correctly, and records are maintained.

4.2.6 One person may fill more than one of these positions from time to time since the project is fairly small. However, during any on-site operations, the Project QA Coordinator will not fill any other job but will report directly to the CFX-PM and, through the EKC QA Representative, to the EKC-PM.

4.2.7 Plans, Procedures and Reports will be generated through the NEXTEP administration and publications department.

4.3 POLICIES AND PROCEDURES

4.3.1 Objective

4.3.1.1 Written policies and procedures governing the decommissioning activities will ensure that a rigorous and standard process is followed and is documented. All facets of the required activities, including radiation safety, instrumentation and calibration, and work processes for each area will be referenced and followed.

4.3.1.2 Because the decommissioning work will be performed primarily by persons familiar with radiological and decommissioning activities, use of the personnel, procedures, instrumentation, and calibration routines currently in use will be applied to the maximum extent practicable at EKC.

4.3.2 Responsibilities

4.3.2.1 All employees working on the decommissioning project are required to understand and follow specialized written procedures and review and acknowledge special work permits for each specific job. During decommissioning activities, all employees are to bring to their supervisor's or manager's attention any circumstance that is different from what is called for in the work plan/special work permit or could present an unsafe situation or cause unnecessary radiation exposure.

4.3.2.2 The project management team is to assure that the proper tools and procedures are available to do the work assigned and to be responsive to any unsafe or unusual exposure situations.

4.3.3 Written Policies, Procedures and/or Work Permits

4.3.3.1 All work performed during decommissioning activities is to be in accordance with written directives that have been reviewed and approved by the decommissioning management, QA and RSO's. Any deviations from the written guidance are to be identified and accompanied with the appropriate documentation that the change was noted, discussed, and approved by the RSOs and decommissioning management.

4.3.4 Worker Training

4.3.4.1 The personnel that will be associated with the decommissioning effort typically have specialized experience in the activities to be performed. Training programs, specific to the needs of workers and EKC employees that may require access to certain areas while decommissioning is going on, will be utilized. Training of workers will be documented, and routine Special Work Permit (SWP) training will be given before any remediation or final status survey task is begun.

4.3.5 Safety Policies and Procedures

4.3.5.1 All work is to be done in a manner that is safe and in accordance with both EKC and NEXTEP safety policies and procedures. NEXTEP personnel are required to attend EKC Contractor Health, Safety and Environment training prior to starting work. In a circumstance where there is a potential conflict between EKC and NEXTEP procedures, there will be a meeting of decommissioning management and the RSO representatives to resolve the matter before proceeding. A written record will be maintained for any such decision.

4.3.6 Applicability of EKC Policies and Procedures

4.3.6.1 Eastman Kodak Company policies and procedures hold precedence over day to day activities at the EKC Research Laboratories. Specific CFX and Building 82 decommissioning policies and procedures hold precedence with regard to decommissioning activities at EKC. It is the responsibility of management to assure that there is as much consistency as possible.

4.3.7 Contingency Situations

4.3.7.1 In case of a non-radiological emergency or accident, EKC procedures will be followed. The CFX Decommissioning procedures will provide direction in case of any conceivable radiological incident and those procedures will take precedence in the response.

4.4 QA/QC RESPONSIBILITIES

4.4.1 The objective of the QA/QC function is to ensure that data are collected and work is done in accordance with applicable procedures, that license conditions are followed, and that complete records of activities are maintained. Additionally, the QAC will review all survey data in accordance with procedures and he will ensure that the data meet the requirements for use.

4.4.2 Primary QA/QC responsibility for the CFX Decommissioning Project rests with the Project QA Coordinator. The QAC will report directly to the CFX-PM and, through the EKC QA Representative, to the EKC-PM.

4.4.3 The EKC-PM or the EKC QAR may request an audit of the project at any time. The independent audit will generate a written report noting areas of compliance and deviation from policies, procedures, and license conditions and is to be submitted to the EKC-PM. The EKC QAR will assure that a written response to the independent audit is prepared and that corrective actions have been completed. The audits and corrective actions taken are to be kept on file and available for NRC during their inspections.

4.5 NRC/STATE OF NEW YORK

4.5.1 NRC Site Inspections

4.5.1.1 The NRC, at any time, may inspect the facility and observe the activities occurring to achieve the decommissioning goal, ensure that licensee activities are in compliance with NRC rules and license conditions, and comment on planned activities. Inspections may be announced or unannounced.

4.5.2 Coordination with the State of New York

4.5.2.1 Copies of materials submitted to the NRC will be provided to the State of New York. Conversely, any letters from the licensee to the State of New York relating to the decommissioning will be provided to the NRC.

5 METHODS TO PROTECT HEALTH AND SAFETY

5.1 RADIATION PROTECTION PROGRAM

5.1.1 The facility will utilize an established, written, Radiation Protection Program to assure there are formal responsibilities and methods for detecting radiation and keeping exposures to individuals "as low as reasonably achievable" (ALARA) during decommissioning activities. Oversight of these programs is provided by the EKC RSO.

5.1.2 Personnel Protection

5.1.2.1 For decommissioning work involving decontamination, workers will wear personnel protective equipment in accordance with facility health and safety (including OSHA) and Radiation Protection Program requirements. The only work anticipated inside the building is associated with surveys for final release. Should a significant amount of contamination be located, workers will wear personnel protective equipment as needed to comply with applicable special work permits (SWPs) and facility health and safety requirements.

5.1.2.2 For decommissioning work associated with the removal of materials and incidental contamination associated with activation, decommissioning workers will wear coveralls, safety shoes, and safety glasses with side shields, and gloves, at a minimum when materials are physically being removed.

5.1.2.3 If analysis of a decommissioning activity suggests airborne contamination may be generated, the worker will be issued appropriate respiratory protection before the work is undertaken. The need for respiratory protection is addressed via the SWP documenting that the potential hazard has been evaluated (i.e., ALARA considerations) and proper precautions against unnecessary exposure have been taken.

5.1.3 Personnel Monitoring Devices

5.1.3.1 All individuals working with radioactive materials are required to wear a film badge. These badges are worn on the outside of the protective clothing between the neck and waist. The badges are exchanged at least quarterly and the individual's dose for the period recorded. The EKC RSO maintains records of employee exposures on file at the EKC Research Laboratories and they are available for both employee and NRC review.

5.1.4 Work Area Monitoring Devices

5.1.4.1 Air sampling may be performed if extensive remediation is warranted.

5.1.5 Ambient Exposure Monitoring

5.1.5.1 Routine micro-R surveys will be performed during decommissioning activities to ensure that regulatory dose limits are met.

5.1.6 Radiation Detection Instruments

5.1.6.1 Table C-1 provides a list of the Radiation Monitoring Instruments available for use during CFX Decommissioning activities. Although additional, equivalent instrumentation may be added to the list, it is adequate to complete the project. If specialized analytical instrumentation is required because of unforeseen levels of detection, samples will be processed at a certified, commercial laboratory off site.

5.1.7 Calibration Standards and Frequency

5.1.7.1 Established written policies and procedures will address the use and calibration of survey equipment to be used for this project. Each instrument is identified by serial number and records of its use and calibration are maintained on file. Calibration of all final release survey instrumentation shall be traceable to NIST standards.

5.1.8 ALARA Committee

5.1.8.1 All decommissioning activities are subject to review and oversight by an established ALARA Committee. The Committee will be comprised, as a minimum, of the EKC-PM, both RSO's, the CFX-PM, and the QAC. The EKC RSO shall chair the committee.

5.1.8.2 This committee will meet prior to commencement of site work and when special circumstances and/or exposures occur. An agenda and record of the ALARA Committee meetings will be maintained by the EKC RSO and be available for management and NRC inspection. The EKC RSO will set the agenda and can also call special meetings to review requests for special exposures or significant changes in procedures.

6 FINAL RADIOLOGICAL STATUS SURVEY

6.1 OVERVIEW

6.1.1 The final status survey will be performed in accordance with guidance contained in NUREG-1757[36] and MARSSIM[37].

6.2 SITE CONDITIONS AT TIME OF FINAL RADIOLOGICAL STATUS SURVEY

6.2.1 At the time of the performance of the Final Radiological Status Survey (FSS), the CFX fuel assembly will have been disassembled and the fuel plates removed and disposed of. In addition, all removable equipment and materials will have been surveyed for free release and disposed of, and all remediation and decontamination efforts will have been completed (if required).

6.2.2 Upon completion of the FSS, the survey documentation will demonstrate that the site meets the criteria for unrestricted release.

6.3 CLASSIFICATION OF AFFECTED AREAS

6.3.1 The source material permitted by Kodak's NRC license is limited to enriched uranium and the products of fission contained within the MTR fuel plates. Since they will have been removed before decommissioning begins, only minimal residual contamination that might be attributed to an undetected breach of the plates during refueling and materials activation within the cavity is expected. A discussion of the radionuclides of interest is presented in Section 3.3.

6.3.2 The areas surveyed during the Characterization Survey are evaluated in the RCR and have been classified as described in Section 3.5 and listed in Table 3.3.

6.3.3 Room C1, the corner of the cavity where the CFX was mounted is classified as Class 2. Although no evidence of any contamination with licensed material was discovered, some residual radioactivity remains in the activated concrete, and it will be evaluated against the limits that are ultimately approved for activated material.

6.3.4 The portion of the cavity surrounding the CFX, room C2, has been classified Class 3. No evidence of either contamination or activation was discovered in this area. Therefore, no activity above background is expected in this survey unit. It will be surveyed because it is adjacent to C1.

6.3.5 The rest of the labyrinth leading out of the cavity to the basement of Building 82 has been classified as unaffected because, like C2, no evidence of either contamination or activation was discovered in this area.

6.3.6 Because the labyrinth and cavity are totally encased in concrete which is up to two feet thick, and because no evidence of contamination has been discovered, no soil or groundwater issues have been considered.

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36 NUREG-1757, Ibid.

37 NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual*

6.4 Data Quality Objectives (DQOs)

6.4.1 The objectives of the Final Status Survey (FSS) and the quality constraints placed upon the data to be generated are as follows:

6.4.1.1 Survey objective: Determine whether the residual radioactivity in the survey unit is less than the release criteria in Section 3 as applicable.

6.4.1.2 The Minimum Detectable Concentration (MDC) for direct measurements should be less than 50% of the relevant release criteria and, if possible less than 10% of the relevant release criteria.

6.4.1.3 The MDC for scans for Class 1 areas should be less than the maximum limit specified in Section 3.

6.4.1.4 The MDC for scans for Class 2 areas should be less than the average limit specified in Section 3. While Kodak's goal during decommissioning is to obtain a scan MDC that is as low as reasonably achievable, is may not always be possible to obtain an MDC less than the limit. In accordance with MARSSIM guidance, any indication of residual radioactivity will be identified for further investigation.

6.4.2 MDC calculations for the instruments used in Characterization and planned for the FSS are discussed in Section 3.10 and the results are presented in Table 3.8.

6.5 Final Status Survey Design Process

6.5.1 Final Status Survey designs will be prepared, reviewed and approved as a written FSS Plan (FSSP) for each survey unit before the survey is conducted. Survey designs will be prepared in accordance with the following Sections.

6.5.2 Identify, Classify, and Describe each Survey Unit.

6.5.2.1 For building surveys, the rooms, hallway, storage areas, etc. to be included in each survey unit will be identified. Every item of fixed apparatus[38] will be identified beforehand as equipment to be surveyed during the FSS. All removable equipment and materials will be released and removed from the survey units prior to FSS.

6.5.2.2 A drawing or sketch of the survey unit will be prepared showing geographic placement of the impacted area or building, room layouts and numbering, location of impacted fixed equipment, survey unit ID, Class designation, and other information as needed. The total surface area of each survey unit will be calculated and recorded in the FSSP and will conform to the criteria in MARSSIM for survey unit size and classification.

6.5.3 Select Representative Background Reference Areas and Materials.

6.5.3.1 Representative reference areas and/or reference materials that correspond to the materials found in the survey units will be selected. At least five measurements of background will be collected on each type of material (matrix) in the reference areas to

[38] Fixed apparatus includes items of equipment which will remain fixed to the building structure and therefore will be released according to building surface criteria. Typical items include ductwork, pipes and conduits, brackets, etc. The terms "fixed apparatus" and "fixed equipment" are used interchangeably in this Plan.

establish a matrix background reference with which to compare survey values during the FSS.

6.5.3.2 The mean and standard deviation of matrix-specific background data will be calculated and listed in the FSS Report (FSSR). Materials for which no background reference measurements can be obtained will be assumed to have background values equal to the LLD for the applicable instrumentation for purposes of the FSS[39].

6.5.4 Prescribe the Survey Methods and Instruments

6.5.4.1 The type of radiation to be measured and the type of instrument to be used will be stated along with measurement techniques and the survey count rate necessary to ensure that the DQO for the Minimum Detectable Concentration (MDC) is met.

6.5.5 Specify the Reference Coordinate System.

6.5.5.1 An unambiguous coordinate system will be adopted which can identify all measurements to their location on any building surface or on any fixed apparatus. The coordinate system for the building surfaces will identify each data point by building, room, room surface (i.e. floor, wall, etc.) and X and Y coordinates measured in feet from a specified origin.

6.5.5.2 Affected fixed equipment located within each room at the time the Final Status Survey is performed will be identified. Survey locations associated with the statistical grid pattern that fall on fixed equipment will be assigned to the surface beneath or behind the fixed equipment and it will be noted that the point fell on the fixed equipment.

6.5.5.3 Additional "bias" measurements that are taken on fixed equipment will be numbered and identified on the associated drawings as well as in the remarks section of the data collection forms.

6.5.6 Specify the Sample Collection Procedures

6.5.6.1 Measurements will be collected and analyzed in accordance with existing procedures and the FSSP.

6.5.7 Determine the Systematic Grid Spacing

6.5.7.1 Measurements will be taken on a regular grid pattern using the criteria for grid spacing contained within MARSSIM. In most instances, the number of data points required will be calculated using the Visual Sample Plan (VSP) program[40]. This program enables the user to input the Type I and II error rates, the estimated standard deviation and the LBGR[41]. The VSP program then automatically calculates the number of samples required and graphs the power curve for both the Wilcoxon Rank Sum (WRS)

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39 Section 3.5

40 The purpose of Visual Sample Plan is to provide simple tools for defining an optimal, technically defensible sampling scheme for characterization. VSP is applicable for any two-dimensional sampling plan including surface soil, building surfaces, water bodies, or other similar applications. It has been developed at Pacific NW Laboratories with support from DOE, DOD, and EPA.

41 "Lower Bound of the Gray Region", a statistical term described in NUREG 1575 (MARSSIM)

Test and the Sign Test. The VSP program operates using MARSSIM techniques and has been validated for use on this project.[42]

6.5.7.2 Due to the fact that there may be some missing or unusable data for a given statistical survey, the rate of missing or unusable measurements, R, expected to occur in a survey unit or reference area, will be accounted for during survey planning. In order to ensure that a sufficient number of data points are collected to attain the desired power level with the statistical tests and allow for possible lost or unusable data, the number of data points required will normally be increased by 20% (R=0.2), and rounded up, as practicable. In the event it is not practical to collect $1.2 \cdot n$ measurements, as few as n measurements will be acceptable without verification by a retrospective power curve.

6.5.7.3 The required number of measurements initially determined for a survey unit may also exceed reasonable bounds. In that event, the design process may be repeated utilizing more suitable input values. Recalculation of the power of the test should be performed following modifications to critical parameters in order to ensure that the statistical tests continue to meet survey objectives.

6.5.7.4 In the event that a Class 1 survey unit area is considerably less than 15 m^2 or a Class 2 survey unit area is considerably less than 100 m^2, and the number of measurements required to satisfy the selected statistical test is unreasonably large, the measurement density may be adjusted to at least one measurement per square meter for a Class 1 survey unit or at least one measurement per 10 m^2 for a Class 2 survey unit.

6.5.7.5 Once the required number of statistical survey points have been determined for a given survey unit, they will be arranged in either a rectangular or triangular grid pattern beginning with a randomly selected survey point. Grid spacing will be determined from the following equations:

$$L = \sqrt{\frac{A}{n}} \qquad \text{Rectangular Grid}$$

$$L = \sqrt{\frac{A}{0.866 * n}} \qquad \text{Triangular Grid}$$

where A is the total area of the survey unit and n is the number of survey measurements required in the survey unit.

6.5.8 Determine Bias Data Collection Requirements

6.5.8.1 In addition to the systematic grid survey points, additional "bias" survey points will be taken in areas where radioactivity is likely to be found.

6.5.8.2 The survey designer or survey technician will estimate the number of bias survey points required, especially in and around items of fixed equipment, using process knowledge, existing data and health physics "good practices." Bias survey points may be specified in advance in the FSSP or may be left to the discretion of the survey technician in the field.

[42] NEXTEP TM00-08, *Verification and Validation of Applicable Portions of VSP Software,* A.H. Thatcher, CHP.

6.5.8.3 The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by taking bias measurements at all traps and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior.

6.5.8.4 Structural surfaces, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the building release limits.

6.5.9 Specify Beta/Gamma Scan Measurements

6.5.9.1 The FSSP will specify the extent to which scans will be performed in each survey unit for the purpose of locating elevated areas not detected by the grid and/or bias sample measurements. Beta/gamma scans will be performed using scintillation and/or gas proportional detectors. Directions for scanning (i.e. scan rate and elevation) will be described in the FSSP and will include the following:

- Class 1 Areas. Surface scans will be performed over 100% of the survey unit surfaces and fixed equipment.
- Class 2 Areas. Surface scans will be performed on at least 10% of structure surfaces and/or fixed equipment.

6.5.9.2 The FSSP will contain a floor plan drawing for each survey unit. In the field during the FSS, scan locations will be specified on the drawing and identified by letters corresponding to data entered on the survey form.

6.5.10 Specify Contingency Action

6.5.10.1 If the scan threshold specified in the FSSP is exceeded in the field, the survey technician will confirm the high scan reading with direct β/γ readings and will attempt to determine the size of the impacted area. All scan readings in excess of the scan threshold specified in the FSSP will be confirmed with direct readings and/or samples.

6.5.10.2 Survey forms used for the collection of the data in the field will be included in the FSSP. Normally these survey forms will be filled out in the field, reviewed by the Quality Assurance Coordinator or designee, and forwarded to the data analyst assigned to review the data.

6.5.10.3 When direct readings confirm that radioactivity exceeding the limits exists in a class 2 survey unit, the elevated area will be reclassified as Class 1 and the FSSP will be adjusted to reflect the change. Survey unit areas in the vicinity of the elevated reading will also be evaluated to determine if any additional measurements are needed and whether reclassification is necessary. The survey designer and/or analyst will provide direction regarding additional characterization and/or remediation requirements. The actions taken and the evaluation of the final results will be documented in the FSSR.

6.6 FINAL STATUS SURVEY INSTRUMENTATION

6.6.1 The instrumentation utilized to generate the FSS data will be maintained in accordance with Kodak Radiation Protection Program procedures. These procedures utilize the guidance contained in ANSI N323-1978[43]. Requirements include traceability of calibrations to NIST standards, field checks for operability, background radioactivity checks, operation of instruments within established environmental bounds, training of individuals, scheduled performance checks, calibration using isotopes of energies similar to those to be measured, quality assurance tests, data review, and record-keeping.

6.6.2 Portable survey instruments will be calibrated annually. Where applicable, activities of sources utilized for on-site calibration will be corrected for decay. In addition to the periodic calibration requirements, source response checks will be performed on a daily basis for all instruments being utilized in FSS work.

6.6.3 If an instrument fails a source check, all data collected prior to the source check, up to and including the last satisfactory source check, will be reviewed. Acceptance of the questionable data will hinge upon whether the instrument failure would result in over-reporting or under-reporting. While either case might result in discarding the data, the case of under-reporting is more critical since it could affect decisions with health and safety implications. In all cases where the source check failure could result in under-reporting, data will be further scrutinized and quality control measurements will be used as necessary to assist the evaluator in the decision to either accept or reject data.

6.6.4 All calibration and source check records will completed, reviewed, signed-off and retained in accordance with the Kodak Quality Assurance Program and the CFX Decommissioning Project Work Plan.

6.6.5 A complete listing of the instrumentation available to the CFX program is presented in Table C-1. Specification sheets for the beta and gamma probes used are also included in Appendix C.

6.7 EVALUATION OF FINAL STATUS SURVEY RESULTS

6.7.1 The evaluation of the FSS results involves a number of steps that include: review of the DQOs and survey design; initial verification of the data; analysis and validation of the data; response to elevated measurements; the performance of statistical tests; and documentation of results. These steps are described below and provide the framework that allows for a complete assessment of the survey data, provides an investigation mechanism for problems encountered in survey units, and enables the user to draw final conclusions that are well documented.

6.7.2 **Review and Approval of the Survey Design**

6.7.2.1 All Final Status Survey Plans used in the CFX Decommissioning Project will be prepared in accordance with this Decommissioning Plan (specifically Section 6) as approved by the NRC. Each FSSP will be reviewed and approved prior to being implemented.

[43] ANSI N323-1978, *Radiation Protection Instrumentation Test and Calibration*

6.7.2.2 Specific items to be checked during the review process are:

- Verify the survey design and ensure the number of samples obtained are appropriate.
- Verify whether the DQOs have been modified from the default values. If so, then ensure appropriate justification exists for any departure.
- Verify that the direct measurement MDC is less than 50% of the release limit.
- Verify that the scan MDC is less than maximum limit for Class 1 survey units.
- Verify that the scan MDC is less than average limit for Class 2 or Class 3 survey units or that appropriate investigations are performed when the limit is exceeded.

6.7.2.3 FSSP's are classified as controlled documents and any changes will be issued as approved revisions.

6.7.3 Initial Data Verification

6.7.3.1 Data verification involves the comparison of the collected data with the prescribed activities documented in the FSSPs. All data collected in the field will be submitted to the Quality Assurance Coordinator (QAC) for initial review and forwarding.

6.7.3.2 The QA Coordinator is responsible for the initial screening of the incoming data to ensure that FSSP requirements have been fulfilled and that the data have been legibly and accurately recorded. This review should include the following:

- Evaluate the completeness of the forms and data tables.
- Verify records of instrument calibration.
- Verify records of survey technician training qualifications.
- Make an assessment of the overall quality of the data. This should involve a check to identify gross errors in data recording.
- Verify that the data sheets confirm that measurements were obtained in the correct locations per the FSSP.

6.7.3.3 The QAC is also responsible for the accurate transcription of the survey data from paper records into the project's database. Some of the verification required (e.g. calibration of equipment or training of the surveyor) will be performed automatically within the database.

6.7.3.4 After the data has been entered into the database, the QA Coordinator will file the original records in the controlled document files. The QA Coordinator will maintain the files of all FSS data taken in connection with the project. At the conclusion of the project, these files will be turned over to EKC for final archiving.

6.7.4 Data Validation and Evaluation

6.7.4.1 Data Review and Lock. Data validation involves the comparison of the collected data to the documented DQOs. Once the FSS data has been verified, transcribed and filed by the QAC, he will lock the records in the database, a function for which only QA personnel have access.

6.7.4.2 Following initial review and validation, the data will be reviewed by a technical analyst assigned to the project. This examination will determine if every data point taken in the survey unit is suitable for FSS and will check for the following items:

o The number of points taken are in accordance with the specific FSSP requirements. o The location of regular grid points correspond to what was prescribed in the FSSP. o If some locations were inaccessible and the field team substituted additional grid points, this review will assess their adequacy for use in the statistical calculations required by MARSSIM. o Scan data have been adequately processed. Scan data sheets will be reviewed to determine that any high readings have been adequately verified with direct confirmatory measurements. o Bias data points were taken in accordance with FSSP instructions and their locations accurately documented on drawings and in the database.

6.7.4.3 If, during the technical review, any measurements are considered doubtful, additional measurements may be called for, collected, and added to the data package using the same process.

6.7.4.4 Once this has been completed, the data package is deemed an accurate picture of the survey unit as it was initially surveyed.

6.7.4.5 Data Screening Tests. After being "locked," the data set for the survey unit will then be evaluated using screening software for the comparison tests listed in Table 6.1:

Table 6.1

Data Screening Tests
Min/Max Test
Background Screen
Limit Screen
Average Test
EMC Limit Screen

6.7.4.6 Data Screening tests that are labeled "Screen" will compare each individual data point with a specified threshold and will flag the data point if it fails. If any one data point in the survey unit fails the screening test, the survey unit as a whole will also fail the test. Other tests apply only to the survey units as a whole.

6.7.4.6.1 *Min/Max Test*: All the recorded data points in the survey unit will first be evaluated to determine the difference between the largest survey value and the smallest applicable background value. If the difference is less than the average limit[44], a Class 1 or Class 2 survey unit will be classified as clean (i.e. passes the test) and no further computations will be required. A class 3 survey unit that passes will be further evaluated under the Background Screening test. If a Class 2 or Class 3 survey unit fails the Min/Max test, it will be evaluated and recommended for either reclassification, remediation or further analysis. A Class 1 survey unit that fails this test will be further evaluated under the remaining tests.

6.7.4.6.2 *Background Screening*. All Class 3 survey units will be evaluated using this test. Each survey data point will be compared with the background threshold specified in Table 6.2 and, if it fails the test, it will be flagged as an exception. In Class 3 survey units, no residual contamination is expected. Therefore, investigation levels are set to flag

[44] "Limit" or "average limit" as used in this section refers to the maximum permissible average reading taken over one square meter area and corresponds to the DCGLw . "Elevated Measurement Comparison " (EMC) refers to the maximum limit permissible over an area no larger than 100 cm^2. Both limits are specified in Section 3.

measurements that are just above the range expected for background levels or just above detection limits for the measurement method, whichever is greater.

6.7.4.6.3 Because matrix materials with no background reference data are assumed to have a mean value equal to the LLD of the instrument used, and since there may be unpredictable variation in the distribution of background radioactivity within a survey unit (SU), failure of the background threshold screening test does not automatically require a Class 3 SU to be reclassified as Class 2 or higher. In such cases, further statistical analysis will be performed on Class 3 measurements to identify the modality of the distribution and/or its shift from the expected background value. The reclassification decision will be made based upon these and any other relevant factors, and they will be documented in the FSS Report (FSSR).

Table 6.2
Final Status Survey Screening Tests

Data Screen	Building Surfaces & Fixed Equipment
Background	Fail if: Gross $\beta/\gamma >$ max of ($\bar{x}_{ref}{}^{a}$ $+3\sigma_{ref}$) or MDA
Limit	Fail if: Gross $\beta/\gamma >$ (Limit + $\bar{x}_{ref}{}^{a}$ $+ 3\sigma_{ref}$)
EMC[b]	Fail if: Gross $\beta/\gamma >$ (EMC + $\bar{x}_{ref}{}^{a}$ $+ 3\sigma_{ref}$)

[a] $\bar{x}_{ref}{}^{a}$ and σ_{ref} are the mean and standard deviation of the reference or background measurements calculated from measurements.

[b] The value for EMC used for this screening test will be the maximum limit specified in Section 3.

6.7.4.6.4 *Limit Screening*. In Class 2 survey units, measurements of net levels above the DCGLw are not expected. Therefore, investigation levels are set to flag measurements exceeding the average limit on a net basis. All the survey points within the survey unit will be compared with the DCGLw in accordance with Table 6.2. Reference levels for each data point will be background measurements associated with the same Matrix Code. Data points that fail the test will be flagged as exceptions.

6.7.4.6.5 *Elevated Measurement Comparison (EMC) Screening*. For Class 1 survey units, measurements above the average limit are expected, so the maximum (EMC) limit will be used for this test. All the survey points within the survey unit will be compared with the EMC limit in accordance with Table 6.2. Reference levels for each data point will be background measurements associated with the same Matrix Code. All values exceeding EMC criteria will be flagged as exceptions.

6.7.4.6.6 *Average Test*: The average value for all the survey readings will be compared to the average of all the applicable background readings by matrix. If the difference is greater than the release limit, the survey unit fails the test. When a survey unit (Class 1 only) contains individual data points which exceed the limit, the survey unit may still be released provided that it passes this test and the applicable statistical test[45].

6.7.4.7 The screening software will enter an exception code in each record for those data points that fail any of the screening tests. A description of all the exceptions can also be provided by the screening software for subsequent analysis and review, if required.

6.7.4.8 A flow chart which describes the logic behind the application of the above tests is presented in Figure A-6.1.

6.7.4.9 Statistical Tests. Class 1 survey units which fail the Min/Max Test criterion may still be released provided the survey unit can be shown to meet certain statistical tests. The analyst can process the data points in the survey unit by selecting and using the statistical test that is most appropriate to the survey unit from the list in Table 6.3:

Table 6.3

Statistical Tests
Wilcoxon Rank Sum Test (WRS)
Sign Test for Paired Data
Sign Test

6.7.4.9.1 *Wilcoxon Rank Sum (WRS) Test.* Also called the "Two-sample Wilcoxon Rank Sum Test", it should be used when there is background radiation present and the background characteristics are homogeneous for the materials present in the survey unit. The WRS test assumes that the background reference area and survey unit data distributions are similar except for a possible shift in the medians. This test is normally preferred for land areas being evaluated by soil sampling and therefore is not applicable for the CFX Project.

6.7.4.9.2 *Sign Test for Paired Data.* This test should be used when the background materials differ significantly within the same survey unit. This test will be the most common test used within buildings because of the wide variety of matrix materials present.

6.7.4.9.3 *Sign Test.* Also called the "One-sample Sign Test," this test should be used when background is so low as to be undetectable. The Sign test evaluates whether the median of the data is above or below the DCGL. This test is really a special case of the Sign Test for Paired Data where background is defined as zero.

6.7.4.9.4 A detailed description of all the statistical tests can be found in MARSSIM[46], NUREG-1505[47], and in a variety of statistical texts.

6.7.4.10 Provided all additional considerations such as scan data, swipes, sampling of removable contamination or sludge from traps, etc. indicate that the survey unit (SU) meets the release criteria, the release of the survey unit can be determined from the test results according to Table 6.4.

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45 No measurements above the release limit may cover an area greater than 100 cm^2.

46 NUREG 1575, *Ibid.*

47 NUREG-1505, *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys*; superseded by NUREG 1757.

Table 6.4
Requirements for SU Release [48]

Test	Class 1	Class 2	Class 3
Min/Max	Not required[a]	Not required[a]	PASS
Background	Not required	Not required	PASS
$DCGL_w$	Not required	PASS	PASS
$DCGL_{avg}$	PASS	PASS	PASS
EMC	PASS	PASS	PASS
Statistical Test	PASS	PASS	PASS

[a] Class 1 or 2 survey units which pass Min/Max may be released without further consideration.

6.7.4.11 If the survey unit does not pass the applicable tests listed in Table 6.4, remedial action and/or re-sampling may be required and will be performed as necessary.

6.8 DOCUMENTATION OF FINAL STATUS SURVEY RESULTS

6.8.1 When a survey unit passes all required tests for release the data package will be utilized to complete the Final Status Survey Report (FSSR) for the survey unit. The FSSR will describe all results of the FSS and will be submitted to the NRC in conjunction with a license amendment request to release the survey unit from License SNM-1513.

6.9 FINAL STATUS SURVEY ADMINISTRATION AND CONTROL

6.9.1 The Final Status Survey for Kodak will be controlled and administered to ensure that the results are accurate, secure, and available to authorized personnel and that the work is done in a safe and secure work environment. Special Work Permit (SWPs) and Work Plans (WPs) will be developed and approved prior to commencement of the field work required for the FSS and will specify the type of instrumentation to be utilized in performing the FSS. These SWPs and WPs are an integral part of the radiation protection and quality assurance programs.

6.9.2 Project Organization

6.9.2.1 The EKC-PM has management oversight for the entire project on behalf of EKC and will ensure that all policies and procedures either in place or developed for the FSS meet license and/or decommissioning requirements and are compatible with existing company programs.

6.9.2.2 The FSS will be conducted under the direction of the CFX-PM, who will be supported by a team consisting of qualified site and contractor personnel as described in Section 4. All remediation, transportation and survey data collection and evaluation activities will be directed by this manager. He will also be responsible for generating and enforcing all the applicable SWP's and work plans.

6.9.2.3 Data verification will be performed by the Project QA Coordinator who will also be responsible for data entry and maintenance of the authorized data files.

[48] See MARSSIM, Chapter 8, Table 8.2.

6.9.2.4 NEXTEP analysts will be assigned as required to assist the CFX-PM in drafting FSS Plans, validating and evaluating the survey results, making recommendations to the RSO regarding release or remediation, and producing reports for each survey unit.

6.9.2.5 The final FSSR will be produced by NEXTEP admin personnel and forwarded to EKC for submission and acceptance by the NRC.

6.9.3 Training

6.9.3.1 Training will be provided to site personnel and any other personnel (i.e., contractors, visitors, etc.) who are allowed access to site areas undergoing decommissioning. All members of the FSS team will attend an in-house training session on any SWP and WP prior to commencement of work. All applicable FSS plans, procedures and quality assurance requirements will be reviewed during this training session.

6.9.4 Radiation Protection Program

6.9.4.1 EKC will maintain a Radiation Protection Program that meets and/or exceeds all of the applicable regulatory requirements. All final radiological status surveys will be performed in strict compliance with applicable regulatory and internal requirements. Qualified site and/or contractor personnel will exercise appropriate radiation protection precautions throughout the FSS survey process.

6.9.4.2 Independent EKC Corporate audits for regulatory and internal requirements may be conducted at any time at the discretion of the EKC QA Representative and the EKC-PM. Assessments of program effectiveness will also be performed periodically by the CFX Project QAC. Additionally, NRC and NY State regulatory officials may audit or inspect the program at their discretion.

6.9.5 Quality Assurance Program

6.9.5.1 The CFX decommissioning project will operate under the EKC Quality Assurance Program as adapted to the CFX project by the Decommissioning Work Plan and/or FSSP. The project QAC will ensure that the Work Plan and all associated plans and procedures provide measurements that are collected, controlled, and analyzed in accordance with quality controls and independent reviews sufficient to foster confidence in the resulting data accuracy and validity.

6.9.5.2 The data review process described in Section 6.7.4 will confirm that approved QA/QC procedures have been followed. When and if identified, corrections to recognized deficiencies will be reported and documented using a written Corrective Action Request (CAR) process.

6.10 Petition For Unrestricted Use and Termination of License SNM-1513

6.10.1 The petition for unrestricted release and license termination will be submitted in accordance with requirements contained in NUREG-1757 for Group 3 facilities.

7 SCHEDULE OF DECOMMISSIONING ACTIVITIES

7.1 The schedule for decommissioning is presented in Figure 7.1. The project began in the Fall of 2005 with initial research and technical studies to refine the scope and methods of the CFX decommissioning effort. De-fueling and Characterization are now complete and this revised Decommissioning Plan has been submitted for NRC approval along with the Characterization Report and Final Status Survey Plan.

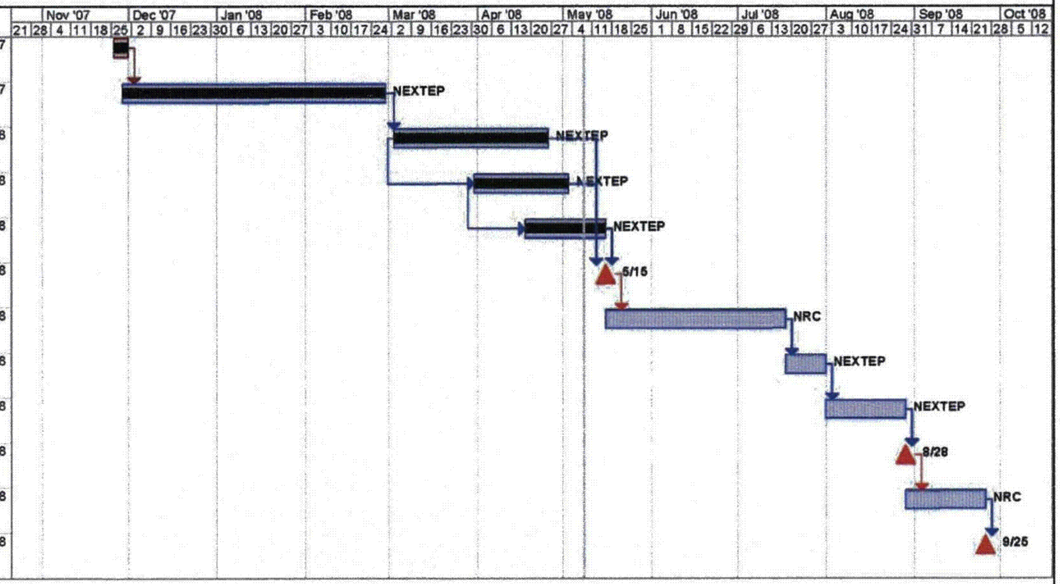
7.2 According to the schedule, the FSSR will be submitted to NRC and NY State Regulators in the third quarter of 2008.

Kodak

Eastman Kodak
CFX Decommissioning Project

CFX Decommissioning Project

ID		Task Name	Duration	Start
1	✓	De-Fueling Field Work	5 days	Mon 11/26/07
2	✓	Characterization	66 days	Thu 11/29/07
3	✓	Draft RCR	8 wks	Mon 3/3/08
4	✓	Draft Dplan Rev. 1	5 wks	Mon 3/31/08
5	✓	Draft FSSP	4 wks	Fri 4/18/08
6		Decom Docs Submitted to NRC/NY	0 days	Thu 5/15/08
7		Review/RAI/Approve Dplan & FSSP	9 wks	Fri 5/16/08
8		Final Status Survey	2 wks	Fri 7/18/08
9		Draft FSS Report	4 wks	Fri 8/1/08
10		FSSR / License Amd Request to NRC	0 days	Thu 8/28/08
11		NRC Review/RAI/Approve FSSR	4 wks	Fri 8/29/08
12		License Release & Completion of Project	0 days	Thu 9/25/08



NEXTEP CONSULTING GROUP

Project Schedule - GANTT

Thu 5/8/08 10:03 PM
080508 - Kodak CFX.mpp

8 DECOMMISSIONING FUNDING

8.1 DECOMMISSIONING ESTIMATE AND SURETY PROVISIONS

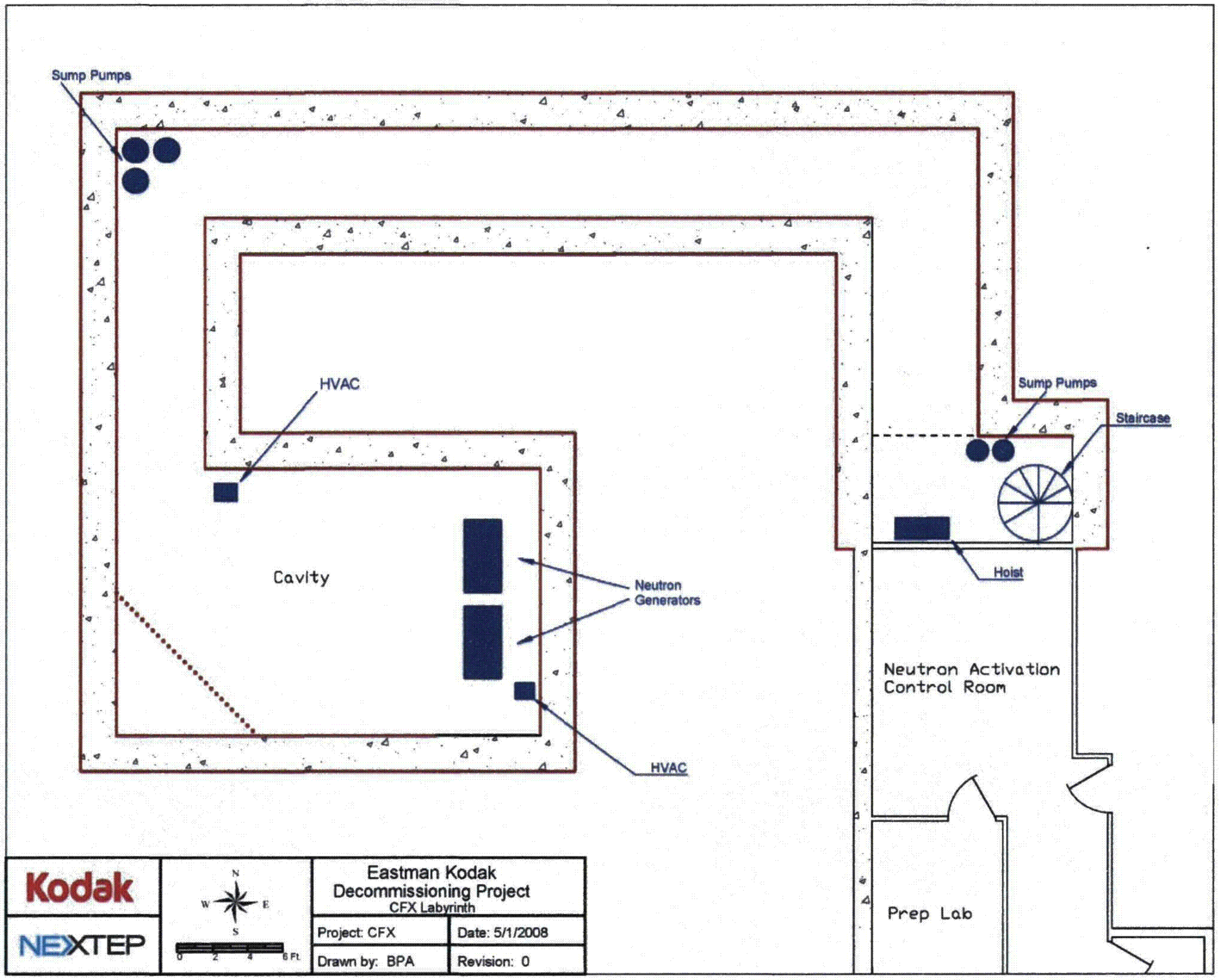
8.1.1 Because the EKC Licensed material (MTR-fuel plates) has all been removed, and given the minor nature of the decommissioning operation, a funding plan is not required for this project by 10 CFR 70.25.[49]

[49] USNRC *Safety Evaluation Report, Application Dated March 23, 1998, Re: License Renewal*

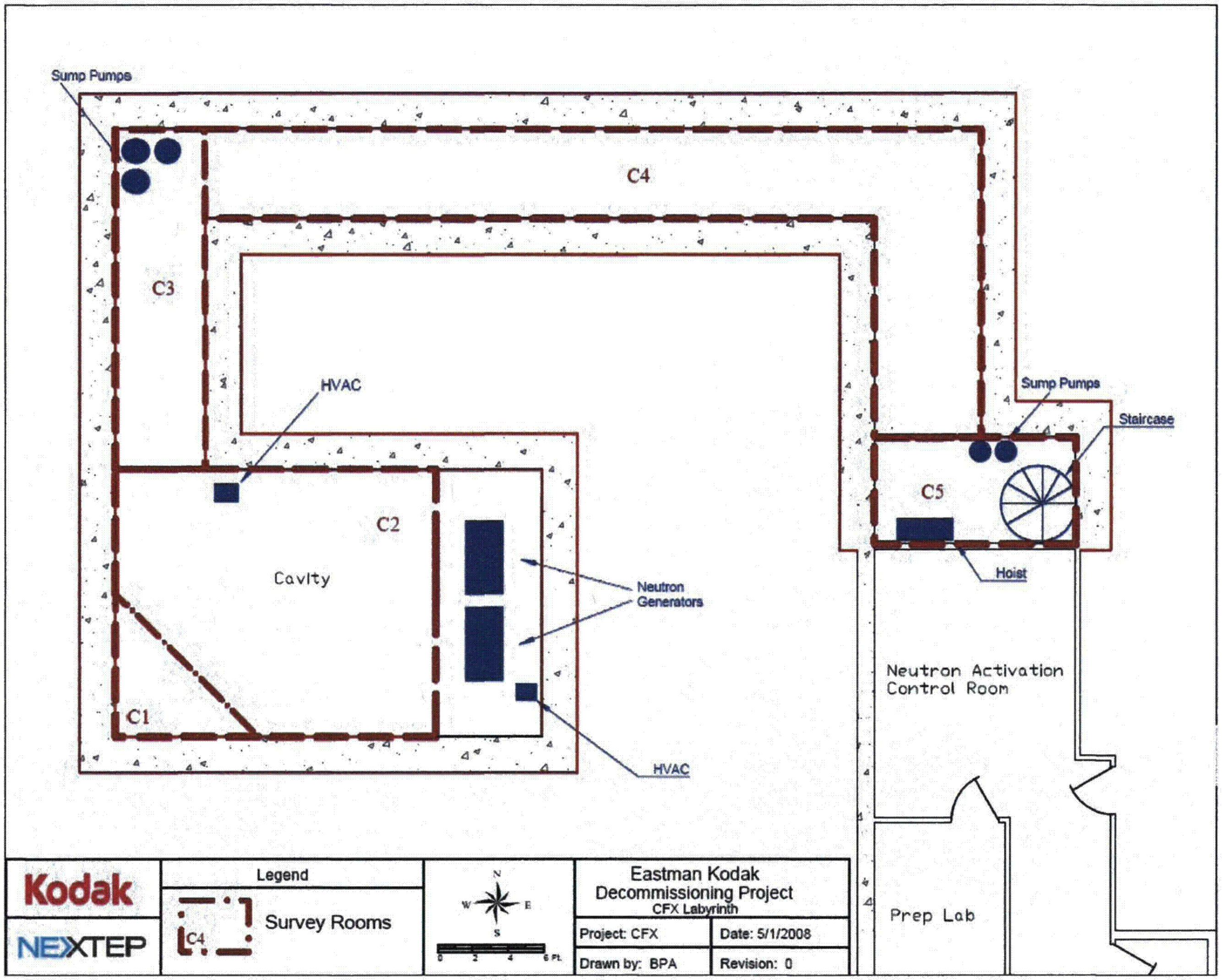
APPENDIX A
Figures

EKC Research Center
Figure A-2.1

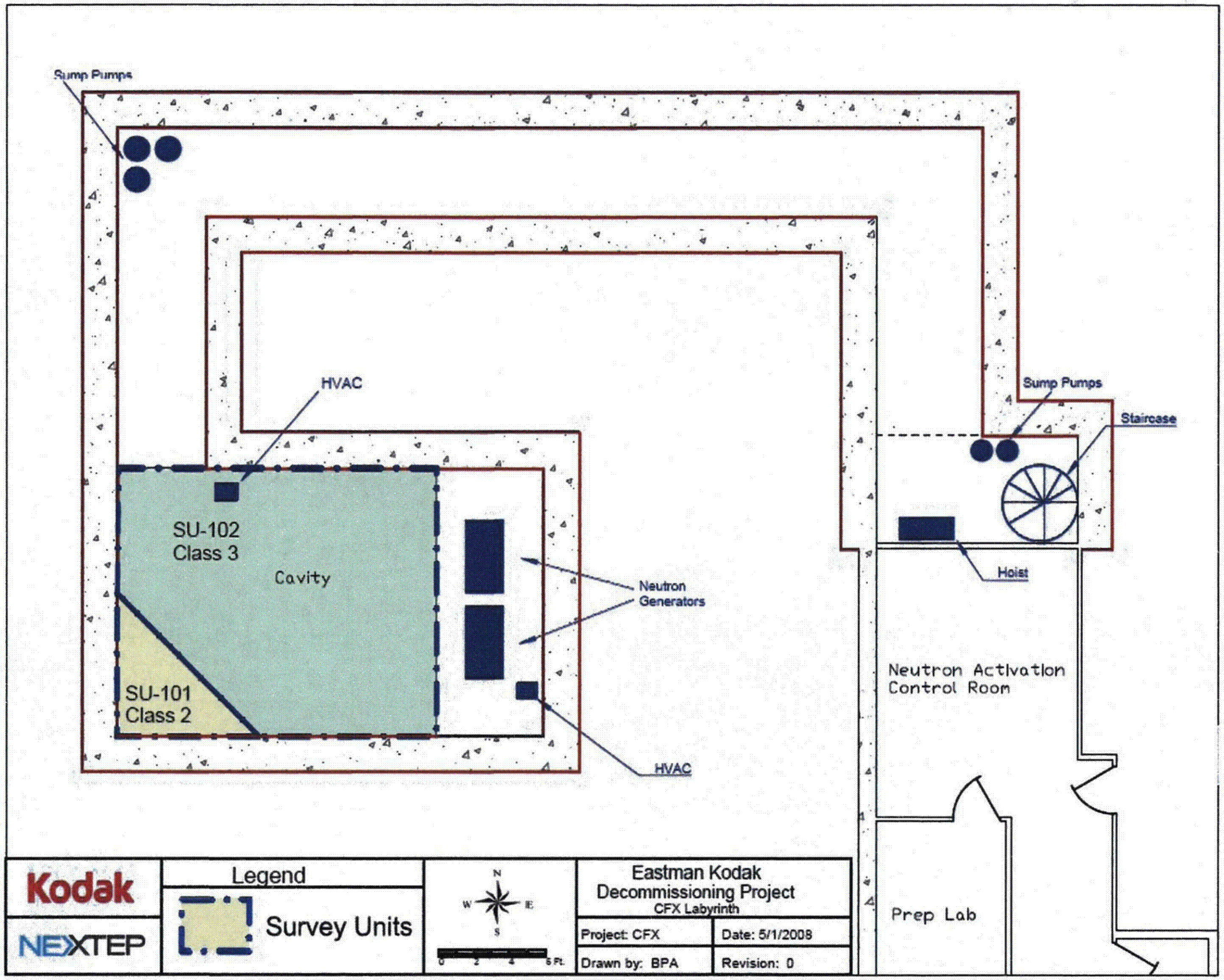
EKC Research Center Buildings
Figure A-2.2



CFX Cavity and Control Room
Figure A-2.3



CFX Constructive Survey Rooms
Figure A-3.1



FSS Survey Unit Classification
Figure A-3.2

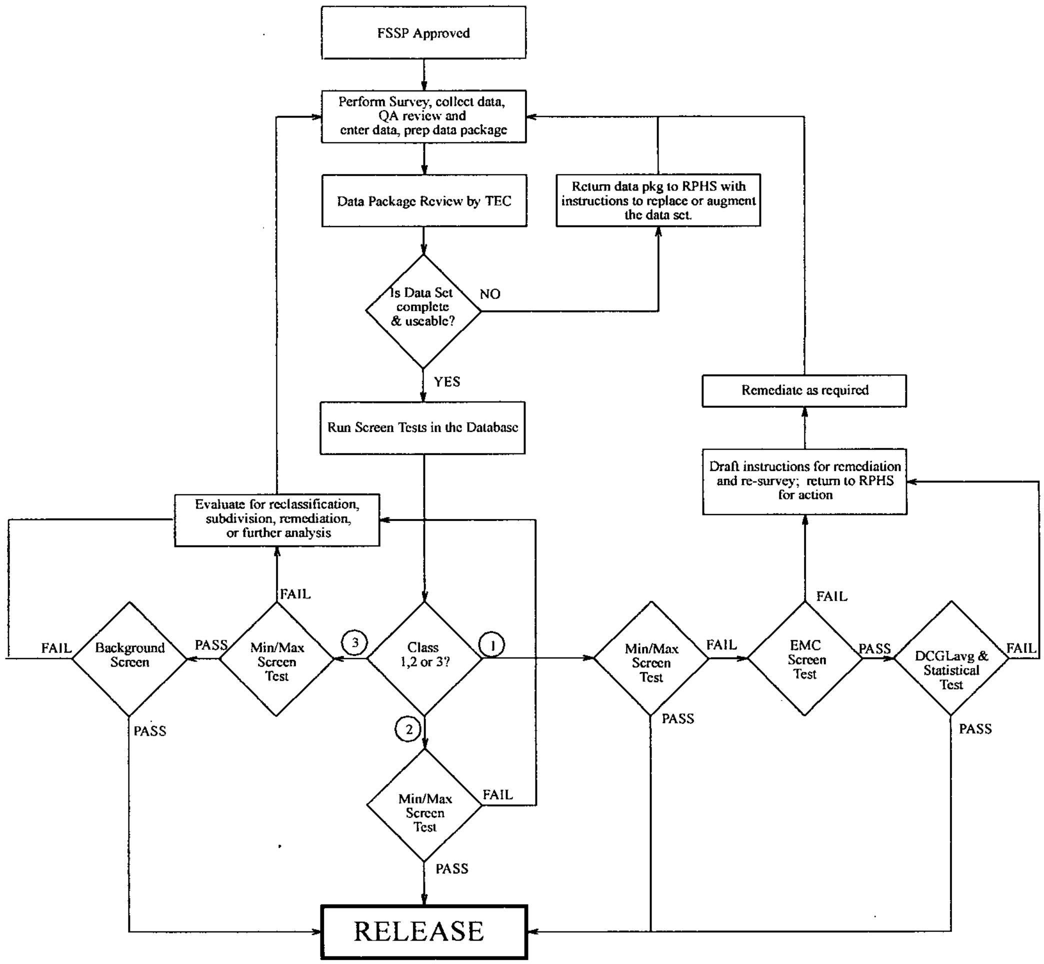


Figure A-6.1
FSS Data Evaluation Process

APPENDIX B
Tables

Table B-3.1
Residual Activity from Actinides (Curies)

Isotope	Curies
tl207	7.61E-07
pb211	7.63E-07
bi211	7.63E-07
po215	7.63E-07
rn219	7.63E-07
ra223	7.63E-07
ac227	7.68E-07
th227	7.55E-07
th231	3.43E-03
th234	3.77E-05
pa231	2.17E-06
pa234m	3.77E-05
u235	3.43E-03
u238	3.77E-05
pu239	5.78E-06
Total	**6.99E-03**

NOTE: Tables taken from NEXTEP TM0519

Table B-3.2
Residual Activity from Fission Products (Adjusted to Benchmark Measurements)

Isotope	Curies
h3	1.64E-04
kr85	4.97E-03
sr89	1.59E-04
sr90	6.41E-02
y90	6.42E-02
y91	3.96E-04
zr95	6.36E-04
nb95	1.37E-03
nb95m	7.46E-06
ru103	2.38E-05
rh103m	2.37E-05
tc99	1.06E-05
ru106	1.58E-03
rh106	1.58E-03
sn123	8.66E-07
cd113m	6.70E-07
sn121	3.30E-07
sn121m	1.50E-06
sb125	3.74E-04
te125m	9.14E-05
te127	1.52E-05
te127m	1.56E-05
te129	1.65E-07
te129m	2.57E-07
cs137	6.61E-02
ba137m	6.24E-02
ce141	1.20E-05
ce144	1.56E-02
pr144	1.56E-02
pr144m	2.18E-04
pm147	2.42E-02
sm151	1.96E-03
eu155	4.90E-04
Total	**3.3 E-01**

Table B-3.3
Beta Background Levels by Matrix

ID	Date	Serial Number	Gamma	Counts Open	Counts Closed	Net Counts	Integrated Minutes	Beta (dpm/100cm²)
Concrete Matrix								
263084	10-Jan-08	162414	2,257	259	200	59	1	1,137
263085	10-Jan-08	162414	2,151	269	171	98	1	1,888
263086	10-Jan-08	162414	2,289	254	181	73	1	1,407
263087	10-Jan-08	162414	2,320	231	199	32	1	617
263088	10-Jan-08	162414	2,275	254	215	39	1	751
			Sum	1,267	966	301	5	5,800
Es =	*0.300*		*Max*	269	215	98		1,888
Ei =	*0.173*		*Average*	**253**	193	**60**		**1,160**
			Stdev	14	17	27		513
Vinyl Tile Matrix								
263089	10-Jan-08	162414	2,128	176	185	(9)	1	(173)
263090	10-Jan-08	162414	2,194	196	164	32	1	617
263091	10-Jan-08	162414	2,196	219	193	26	1	501
263092	10-Jan-08	162414	2,172	205	192	13	1	250
263093	10-Jan-08	162414	2,142	204	235	(31)	1	(597)
			Sum	1,000	969	31	5	597
Es =	*0.300*		*Max*	219	235	32		617
Ei =	*0.173*		*Average*	**200**	194	**6**		**119**
			Stdev	16	26	26		502

Table B-3.4

License Release Criteria
(from License SNM-1513, Attachment 2)

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES (1)	AVERAGE (2, 3, 6)	MAXIMUM (2,4,6)	REMOVABLE (2,5,6)
U-nat, U-235, U-238, and associated decay products	5,000 dpm a/100 cm2	15,000 dpm a/100 cm2	1,000 dpm a/100 cm2
Transuranics, Ra-226, Ra-228, Th-230, Th-22, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm2	300 dpm /100 cm2	20 dpm/100 cm2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm2	3000 dpm/100 cm2	200 dpm/100 cm2
Beta-gamma-emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm by/100 cm2	15,000 dpm b/100 cm2	1000 dpmby/100 cm2

(1) Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

(2) As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

(3) Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

(4) The maximum contamination level applies to an area of not more than 100 cm2.

(5) The amount of removable radioactive material per 100 cm2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

(6) The average and maximum radiation levels associated with surface contamination resulting from beta-gamma-emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

APPENDIX C
Radiological Instrumentation

TABLE C-1
Radiation Monitoring Instruments

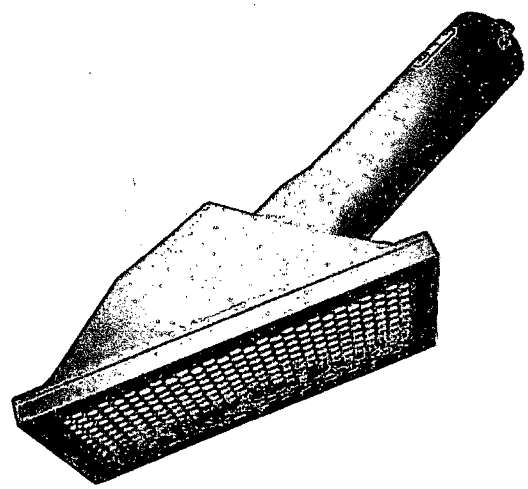
Instrument Type	Number Available	Radiation Detected	Scale Range	Background	Typical Efficiency	Typical MDA @ 95% Confidence
Scintillation Probe (Ludlum 2224/43-89)	2[a]	Alpha Beta	0-500,000 cpm	< 10 cpm < 300 cpm	18% 17%	100 dpm/100 cm^2 500 dpm/100cm^2
Micro-R Meter (Ludlum 19)	3[a]	Gamma	0 – 5,000 μR/hr	7 μR/hr - 9 μR/hr	N/A	2 μR/h
Ion Chamber (Victoreen PIC M450P)	1	Gamma	0.01 mR/hr – 1 R/hr		N/A	
2" x 2" NaI gamma Scintillator (2221/44-10)	2[a]	Gamma	0 - 500,000 cpm	3,000 cpm avg. shielded	N/A	910 cpm (Shielded)
Reuter-Stokes PIC Model RSS-112	1[a]	Gamma	0 – 100 mR/hr	9 – 10 μR/hr	N/A	0.5 μR/h (10min. count)
Geiger Mueller (Ludlum M3/44-38)	1	Beta-Gamma	0.01mR/hr-5R/hr		N/A	
Pancake GM (LudlumM3/44-9)	1	Alpha-Beta	20-100,000 cpm			
RADOS RAD-60 Digital Dosimeters	2	Gamma	0.50-300 mR/hr			
Liquid Scintillation Counter	1	Beta-Gamma	Low E beta (H3)			
Tennelec Series 5 low background GPD w Eclipse software v 3.00	1	Gross alpha/beta			35.78%α/50.07%β	0.76/0.54 dpm α/β
Ortec ultra-low background, high efficiency Germanium gamma spectrometer system	1	Gamma spec				

[a] (Available from Nextep Consulting Group, Inc.)

MODEL 43-89 Alpha/Beta Scintillator

PART NUMBER:47-2430

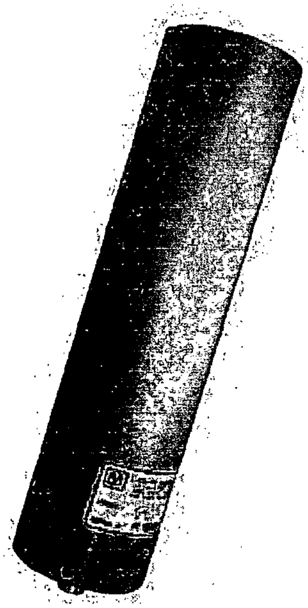
The Model 43-89 is a 100 cm^2 dual phosphor alpha/beta scintillator that is designed to be used for simultaneously counting alpha and beta contamination



INDICATED USE: Alpha-beta survey
SCINTILLATOR: ZnS(Ag) adhered to 0.010" thick plastic scintillation material
WINDOW: Typically 1.2 mg/cm^2 aluminized mylar
WINDOW AREA:
Active - 125 cm^2
Open - 100 cm^2
EFFICIENCY (4pi geometry): Typically 16% - ^{239}Pu; 5% - ^{99}Tc; 16% - ^{90}S/^{90}Y
BACKGROUND:
Alpha - Less than 3 cpm
Beta - Typically 300 cpm or less (*10 microR/hr field*)
NON-UNIFORMITY: Less than 10%
CROSS TALK:
Alpha to Beta - Less than 10%
Beta to Alpha - Less than 1%
COMPATIBLE INSTRUMENTS: Model 2224, 2360, 2929
TUBE: 1.5"(3.8cm) diameter magnetically shielded photomultiplier
OPERATING VOLTAGE: Typically 500 - 1200 volts
DYNODE STRING RESISTANCE: 100 megohm
CONNECTOR: Series "C" (*others available*)
CONSTRUCTION: Aluminum housing with beige polyurethane enamel paint
TEMPERATURE RANGE: -4° F(-20° C) to 122° F(50° C)
May be certified for operation from -40° F(-40° C) to 150° F(65° C)
SIZE: 5.5"(13.9cm)H X 4"(10.2cm)W X 12.3"(33cm)L
WEIGHT: 1.5 lb (0.7kg)

MODEL 44-10 Gamma Scintillator

PART NUMBER:47-1540



The Model 44-10 is a 2" X 2" NaI(Tl) Gamma Scintillator that can be used with several different instruments including survey meters, scalers, ratemeters, and alarm ratemeters

INDICATED USE: High energy gamma detection
SCINTILLATOR: 2" (5.1 cm) diameter X 2" (5.1 cm) thick sodium iodide (NaI)Tl scintillator
SENSITIVITY: Typically 900 cpm/microR/hr (*^{137}Cs*)
ENERGY RESPONSE: Energy dependent
COMPATIBLE INSTRUMENTS: General purpose survey meters, ratemeters, and scalers
TUBE: 2"(5.1cm) diameter magnetically shielded photomultiplier
OPERATING VOLTAGE: Typically 500 - 1200 volts
DYNODE STRING RESISTANCE: 60 megohm
CONNECTOR: Series "C" (*others available*)
CONSTRUCTION: Aluminum housing with beige polyurethane enamel paint
TEMPERATURE RANGE: -4° F(-20° C) to 122° F(50° C)
May be certified to operate from -40° F(-40° C) to 150° F(65° C)
SIZE: 2.6" (6.6 cm) diameter X 11" (27.9 cm)L
WEIGHT: 2.3 lb (1.1kg)